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## Annual Report of Tank Waste Treatability

R. A. Karnesky

Date Published  
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Hanford Company**

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ANNUAL REPORT OF TANK WASTE  
TREATABILITY STUDIES

R. A. Karnesky

ABSTRACT

*This report has been prepared as part of the Hanford Federal Facility Agreement and Consent Order\* (Tri-Party Agreement) and constitutes completion of Tri-Party Agreement Milestone M-04-00. This report provides a summary of treatment activities for newly generated waste, existing double-shell tank waste, and existing single-shell tank waste.*

*Historical reviews of the process by which the grout waste form and the glass waste form were chosen are provided. A compilation of historical reports and abstracts is provided.*

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*\*Hanford Federal Facility Agreement and Consent Order, Washington State Department of Ecology, Olympia, Washington; U.S. Environmental Protection Agency, Region 10, Seattle, Washington; and U.S. Department of Energy, Richland, Washington (May 1989).*

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## CONTENTS

1.0	INTRODUCTION . . . . .	1-1
1.1	TRI-PARTY AGREEMENT . . . . .	1-1
1.2	MILESTONE M-04-01, WORK SCOPE LETTER . . . . .	1-1
1.3	MILESTONE M-04-00, ANNUAL TANK WASTE TREATABILITY REPORT . . . . .	1-1
2.0	SUMMARY . . . . .	2-1
2.1	HISTORICAL OVERVIEW OF TANK WASTE TREATMENT . . . . .	2-1
2.2	TREATMENT OF NEWLY GENERATED WASTE . . . . .	2-1
2.3	PLANNED TREATMENT OF EXISTING DOUBLE-SHELL TANK WASTE . . . . .	2-1
2.4	PLANNED TREATMENT OF EXISTING SINGLE-SHELL TANK WASTE . . . . .	2-2
2.5	BASIS FOR SELECTION OF GLASS AND GROUT DISPOSAL FORMS . . . . .	2-2
3.0	TREATMENT OF EXISTING DOUBLE-SHELL TANK WASTES . . . . .	3-1
3.1	INTRODUCTION . . . . .	3-1
3.2	PLANNED TREATMENT OF DOUBLE-SHELL SLURRY FEED AND DOUBLE-SHELL SLURRY . . . . .	3-1
3.2.1	Definition of Double-Shell Slurry Feed and Double-Shell Slurry . . . . .	3-1
3.2.2	Planned Treatment of Double-Shell Slurry Feed and Double-Shell Slurry . . . . .	3-1
3.3	PLANNED TREATMENT OF NEUTRALIZED CURRENT ACID WASTE . . . . .	3-2
3.3.1	Definition of Neutralized Current Acid Waste . . . . .	3-2
3.3.2	Planned Treatment of Neutralized Current Acid Waste . . . . .	3-2
3.3.3	Schedule . . . . .	3-4
3.4	PLANNED TREATMENT OF NEUTRALIZED CLADDING REMOVAL WASTE . . . . .	3-4
3.4.1	Definition of Neutralized Cladding Removal Waste . . . . .	3-4
3.4.2	Planned Treatment of Neutralized Cladding Removal Waste . . . . .	3-4
3.4.3	Schedule . . . . .	3-4
3.5	PLANNED TREATMENT OF PLUTONIUM FINISHING PLANT WASTE . . . . .	3-6
3.5.1	Definition of Plutonium Finishing Plant Waste . . . . .	3-6
3.5.2	Planned Treatment of Plutonium Finishing Plant Waste . . . . .	3-6
3.5.3	Schedule . . . . .	3-6
3.6	PLANNED TREATMENT OF COMPLEXANT CONCENTRATE WASTE . . . . .	3-6
3.6.1	Definition of Complexant Concentrate Waste . . . . .	3-6
3.6.2	Planned Treatment of Complexant Concentrate Waste . . . . .	3-8
3.6.3	Schedule . . . . .	3-8
3.7	SUMMARY OF DOUBLE-SHELL TANK WASTE TREATMENT . . . . .	3-8
3.8	EXISTING TANK DOUBLE-SHELL TANK INVENTORY . . . . .	3-10
4.0	TREATMENT OF EXISTING SINGLE-SHELL WASTES . . . . .	4-1
4.1	DESCRIPTION OF SINGLE-SHELL TANK WASTES . . . . .	4-1
4.2	TREATMENT OF SINGLE-SHELL TANK WASTES . . . . .	4-1
4.3	STATUS OF SINGLE-SHELL TANK WASTE STUDIES . . . . .	4-1
4.3.1	Destruction of Complexant Concentrate Waste . . . . .	4-1
4.3.2	Removal of Transuranic Components by the TRUEX Process . . . . .	4-2
4.3.3	The Strontium Extraction Process . . . . .	4-2

CONTENTS (cont.)

4.4	SELECTION OF SINGLE-SHELL TANK WASTE FORM . . . . .	4-2
4.5	LABORATORY STUDIES . . . . .	4-2
4.6	ENGINEERING STUDIES . . . . .	4-3
4.7	BACKGROUND INFORMATION . . . . .	4-3
5.0	SELECTION OF GROUT FOR DISPOSAL OF CERTAIN HANFORD SITE LOW-LEVEL LIQUID WASTES . . . . .	5-1
6.0	EVALUATION AND SELECTION OF BOROSILICATE GLASS . . . . .	6-1
7.0	REFERENCES . . . . .	7-1
8.0	BIBLIOGRAPHY . . . . .	8-1
APPENDIXES:		
A	NEW WASTE GENERATORS' FACILITY DESCRIPTIONS, TYPES OF WASTE, AND WASTE MINIMIZATION. . . . .	A-1
B	EVALUATION AND SELECTION OF BOROSILICATE GLASS AS THE WASTE FORM FOR HANFORD SITE HIGH-LEVEL RADIOACTIVE WASTE . . .	B-1

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# LIST OF FIGURES

3-1	Neutralized Current Acid Waste Pretreatment and Disposal . . . .	3-3
3-2	Neutralized Cladding Removal Waste Pretreatment and Disposal . .	3-5
3-3	Plutonium Finishing Plant Waste Pretreatment and Disposal . . . .	3-7
3-4	Complexant Concentrate Waste Pretreatment and Disposal . . . . .	3-9

# LIST OF TABLES

3-1	Double-Shell Tank Inventory as of February 1990 . . . . .	3-12
4-1	Single-Shell Tank Inventory as of February 1990 . . . . .	4-4
5-1	Chronology for the Selection of the Grout Waste Form for Immobilization of Hanford Site Liquid Low-Level Waste: Important Milestones in the United States . . . . .	5-2
5-2	The 1980 Relative Ranking of Selected Waste Forms for Immobilization of Hanford Site Blended Wastes . . . . .	5-3
5-3	Major Advantages and Disadvantages Associated with the Use of Cementitious Grout as a Fixation Medium . . . . .	5-5

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LIST OF TERMS

ANL	Argonne National Laboratory
CC	complexant concentrate
CRW	cladding removal waste
DOE	U.S. Department of Energy
DSS	double-shell slurry
DSSF	double-shell slurry feed
DST	double-shell tank
Ecology	Washington State Department of Ecology
EDTA	ethylenediaminetetraacetic acid
EPA	U.S. Environmental Protection Agency
ER	Environmental Restoration
FY	fiscal year
GTF	Grout Treatment Facility
HEDTA	hydroxyethylenediaminetriacetic acid
HLW	high-level waste
HWVP	Hanford Waste Vitrification Plant
LLW	low-level waste
NCAW	neutralized current acid waste
NCRW	neutralized cladding removal waste
NEPA	National Environmental Policy Administration (Act)
ORNL	Oak Ridge National Laboratory
PFP	Plutonium Finishing Plant
PNL	Pacific Northwest Laboratory
PRF	Plutonium Reclamation Facility
PUREX	Plutonium-Uranium Extraction (Plant)
SEIS	supplemental environmental impact statement
SREX	strontium extraction
SRL	Savannah River Laboratory
SST	single-shell tank
Tri-Party Agreement	<i>Hanford Federal Facility Agreement and Consent Order</i>
TRU	transuranic
TRUEX	transuranic extraction
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## ANNUAL REPORT OF TANK WASTE TREATABILITY STUDIES

### 1.0 INTRODUCTION

#### 1.1 TRI-PARTY AGREEMENT

The basis for this *Annual Report of Tank Waste Treatability Studies* is the *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) which was established in 1989 between the U.S. Department of Energy (DOE), the U.S. Environmental Protection Agency (EPA), and the Washington State Department of Ecology (Ecology) (Ecology et al.). The Tri-Party Agreement contains two milestones, M-04-00 and M-04-01, that address tank waste treatability.

Milestone M-04-00 requires that reports of tank waste treatability studies be completed annually beginning in September 1990. Milestone M-04-01, with a completion date of December 1989, requires that a letter be provided to Ecology describing the work scope to be included in the September 1990 report.

#### 1.2 MILESTONE M-04-01, WORK SCOPE LETTER

On December 29, 1989, the letter describing the work scope of the tank waste treatability report was sent to Ecology (Freeberg 1989). The letter addressed the scope, organization, and annotated outline for the annual report and successfully completed Milestone M-04-01.

#### 1.3 MILESTONE M-04-00, ANNUAL TANK WASTE TREATABILITY REPORT

This report is the first Tank Waste Treatability Report of an annual series required by Milestone M-04-00. This first report provides an historical perspective of tank waste treatment, describes planned treatment of existing double-shell tank (DST) and single-shell tank (SST) wastes, and provides the technical basis for selection of grout and glass as disposal forms.

Future annual reports will provide descriptions of waste treatment activities in the intervening 12 mo.

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## 2.0 SUMMARY

This first *Annual Report of Tank Waste Treatability Studies* document satisfies Tri-Party Agreement Milestone M-04-00 for fiscal year (FY) 1990.

### 2.1 HISTORICAL OVERVIEW OF TANK WASTE TREATMENT

Since the 1940s, there has been an effort to reduce the volume of waste stored in 149 SSTs and, more recently, in 28 DSTs. During the past 46 yr, hundreds of millions of gallons of tank waste have been treated, resulting in an FY 1990 inventory of approximately 37 Mgal of waste in the 149 SSTs and approximately 22 Mgal of waste in the 28 DSTs.

The first major treatment campaign occurred in the 1950s when SST waste was recycled to recover uranium. Another major campaign was the treatment of SST waste in the 1960s and 1970s to remove strontium and cesium which resulted in a lower-heat and safer storage form.

### 2.2 TREATMENT OF NEWLY GENERATED WASTE

Currently, ten major facilities generate new tank waste:

- 100-N Area
- 300 Area
- 400 Area
- Tank Farms
- Evaporators
- Plutonium Finishing Plant (PFP)
- Plutonium-Uranium Extraction (PUREX) Plant
- B Plant
- S Plant
- T Plant

Treatment of this waste is addressed in Appendix A.

### 2.3 PLANNED TREATMENT OF EXISTING DOUBLE-SHELL TANK WASTE

Existing waste in DSTs will be treated in B Plant to separate DST waste into high-level waste (HLW), transuranic (TRU) waste, and low-level waste (LLW) fractions. The HLW and TRU waste fractions will be vitrified in the Hanford Waste Vitrification Plant (HWVP) for permanent disposal in a national geologic repository.

The LLW fraction will be treated in the Grout Treatment Facility (GTF) to make a solid waste form suitable for near-surface disposal in concrete vaults. The Tri-Party Agreement provides schedules for future HWVP and GTF activity.

#### **2.4 PLANNED TREATMENT OF EXISTING SINGLE-SHELL TANK WASTE**

Existing waste in SSTs is being characterized to enable appropriate treatment options to be developed. This information is needed for a supplemental environmental impact statement (SEIS) leading to a decision on final SST waste disposal.

#### **2.5 BASIS FOR SELECTION OF GLASS AND GROUT DISPOSAL FORMS**

Documentation that supports the selection of the glass waste form for disposal of HLW in a geologic repository is described. Documentation that supports the selection of the grout waste form for disposal of LLW in near-surface concrete vaults at the Hanford Site is described.

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### 3.0 TREATMENT OF EXISTING DOUBLE-SHELL TANK WASTES

#### 3.1 INTRODUCTION

Treatment of existing DST wastes is required before permanent disposal (Augustine 1989). The treatment strategy is to separate DST wastes into three portions: HLW, TRU waste, and LLW.

Treatment of the separated HLW and TRU waste fractions will consist of vitrification in the HWVP before disposal in a geologic repository. Treatment of the LLW consists of solidification in cement-based grout before disposal in near-surface vaults at the Hanford Site.

These treatment processes are in various stages of development as discussed below. The planned treatment activities will be discussed according to the waste types of double-shell slurry feed (DSSF), double-shell slurry (DSS), neutralized current acid waste (NCAW), neutralized cladding removal waste (NCRW), PFP waste, and complexant concentrate (CC) waste.

#### 3.2 PLANNED TREATMENT OF DOUBLE-SHELL SLURRY FEED AND DOUBLE-SHELL SLURRY

##### 3.2.1 Definition of Double-Shell Slurry Feed and Double-Shell Slurry

Many streams that enter DSTs consist of dilute liquids low in radioactivity. These streams are so concentrated by Evaporator 242-A that one more pass through would increase the sodium aluminate concentration past the sodium phase boundary and the stream would solidify when cooled. At this point the waste is called DSSF. When the DSSF is processed through Evaporator 242-A, the DSSF is concentrated past the sodium aluminate phase boundary. The hot slurry is pumped to a DST where it forms solids as it cools. The waste is then called DSS.

##### 3.2.2 Planned Treatment of Double-Shell Slurry Feed and Double-Shell Slurry

The DSSF will be pumped from DSTs to the GTF for treatment and conversion into grout. The DSS will be treated in the same manner, except for one additional treatment step to remove the DSS solids from the DSTs.

Milestone M-01-01 of the Tri-Party Agreement calls for the completion of three grout campaigns of DST waste. One campaign of phosphate-sulfate LLW has been completed. The remaining two campaigns will use DSSF and DSS.

Vaults to hold DSSF and DSS grout are under construction. When the vaults are complete, treatment of DSSF and DSS will begin.

Treatment of DSSF-DSS has been studied in the laboratory as part of the Grout Formulation Program to develop and qualify grout formulae for the solidification of the Hanford Site's DST waste. A formula consists of measured quantities of no more than four dry materials (e.g., calcium carbonate, fly ash, blast furnace slag, and cement), no more than three liquid additives, and DSSF or DSS waste.

The dry materials are blended before the liquids are mixed with the waste. Blending refers to the dry materials and mixing refers to the mixing of the dry blend with the waste.

Qualification consists of verifying grout performance as a function of the following expected process variabilities:

- Changes in DSSF and DSS waste composition
- Dry material composition variables
- Changes in dry material storage conditions
- Dry material blending variables
- Variables in the mixing of DSSF-DSS waste with the dry blend
- Variables in grout curing conditions
- Changes in the long-term vault conditions (grout aging).

Studies are being completed to determine the results of these process variables on DSSF and DSS grout.

### **3.3 PLANNED TREATMENT OF NEUTRALIZED CURRENT ACID WASTE**

#### **3.3.1 Definition of Neutralized Current Acid Waste**

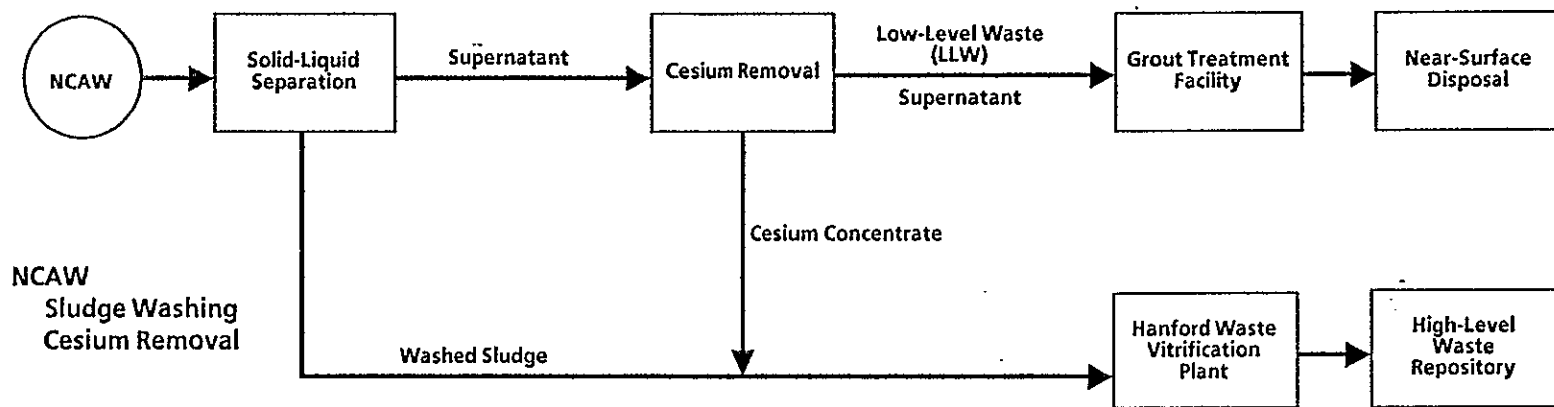
The NCAW is the aqueous high-salt waste from the first-cycle solvent extraction column in the PUREX Plant. This waste is neutralized to prevent corrosion of the carbon steel tanks.

#### **3.3.2 Planned Treatment of Neutralized Current Acid Waste**

The first step in the proposed treatment process is to separate the solids from the supernatant (Figure 3-1) (WHC 1990). Solid-liquid separation is achieved in the laboratory using a settle-decant process (Wong 1989).

The supernatant contains most of the cesium which will be removed by ion exchange leaving a LLW fraction destined for the GTF. Cesium will be eluted from the ion-exchange column and combined with the solids to form the HLW fraction of NCAW destined for the HWVP (Schwoebel and Northrup 1978).

Figure 3-1. Neutralized Current Acid Waste Pretreatment and Disposal.



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### 3.3.3 Schedule

The NCAW treatment technology has been demonstrated in the laboratory. Plant-scale testing in Vault 244-AR and B Plant was scheduled to begin in October 1993; however, as a result of FY 1990 to 1991 funding constraints, the October 1993 date is being revised.

## 3.4 PLANNED TREATMENT OF NEUTRALIZED CLADDING REMOVAL WASTE

### 3.4.1 Definition of Neutralized Cladding Removal Waste

Cladding removal waste (CRW) results from the dissolution of the N Reactor spent fuel Zircaloy cladding using the Zirflex process in the PUREX reprocessing plant. Neutralization of this waste (NCRW) causes most of the zirconium to precipitate as a hydrated oxide, essentially removing all of the actinides and fission products from the solution.

### 3.4.2 Planned Treatment of Neutralized Cladding Removal Waste

The first step in the proposed treatment process is to separate the solids from the supernatant (Figure 3-2). The supernatant is a LLW that can be sent to the GTF for further treatment (Kurath and Yeager 1987).

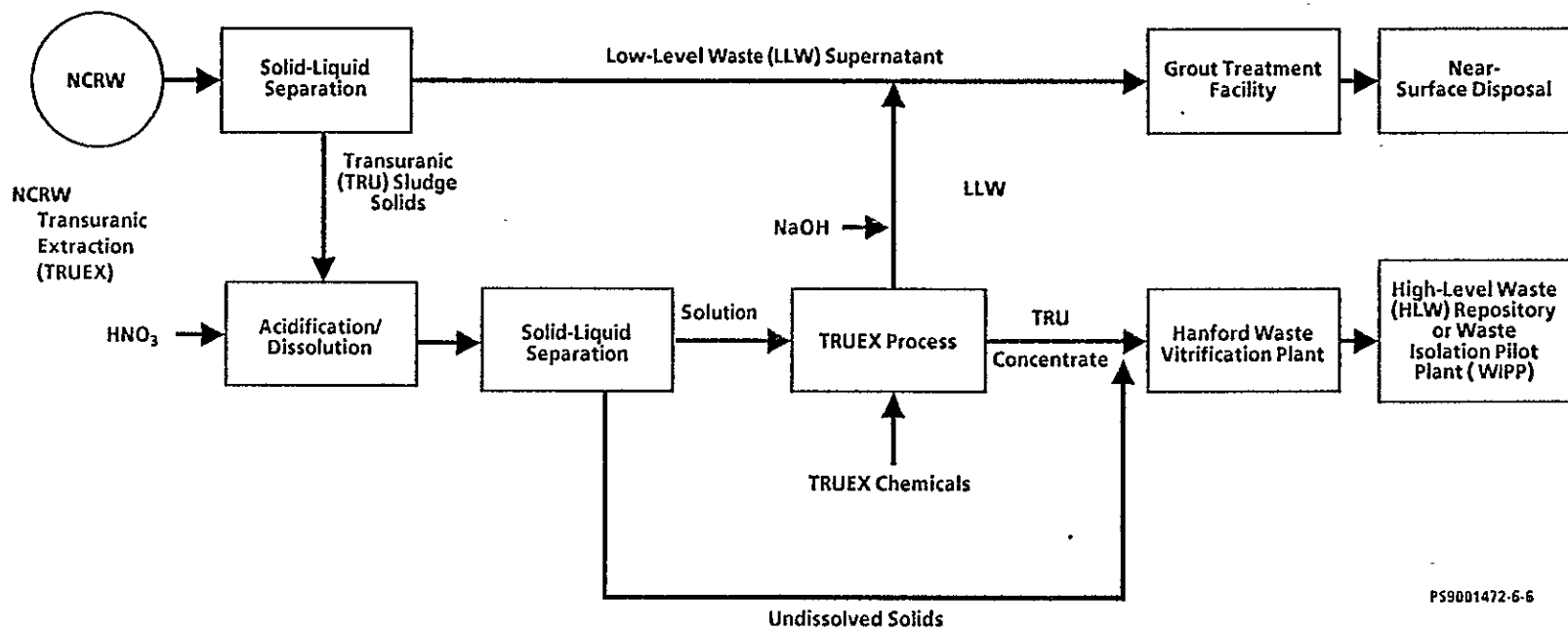
Most of the solids are dissolved with nitric acid. The dissolved TRU elements are separated from the undissolved solids and are used as feed for the transuranic-extraction (TRUEX) process.

The TRUEX process separates a small volume of the concentrated TRU waste from a large-volume LLW stream, the latter being sent to the GTF. The concentrated TRU stream is recombined with the undissolved solids for transfer to the HWVP for vitrification.

### 3.4.3 Schedule

Pilot plant tests with NCRW are scheduled through FY 1993. Operation of the TRUEX process at B Plant using a NCRW feed could commence as early as FY 1998 (Augustine 1989).

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Figure 3-2. Neutralized Cladding Removal Waste Pretreatment and Disposal.

### 3.5 PLANNED TREATMENT OF PLUTONIUM FINISHING PLANT WASTE

#### 3.5.1 Definition of Plutonium Finishing Plant Waste

The PFP waste originates from the conversion of plutonium nitrate to oxide or metal and includes TRU laboratory wastes. The PFP waste also includes Plutonium Reclamation Facility (PRF) waste consisting of high-salt solvent extraction waste and organic wash waste.

#### 3.5.2 Planned Treatment of Plutonium Finishing Plant Waste

The first step in the proposed treatment process is to separate the solids from the supernatant (Figure 3-3). The supernatant is a LLW that can be sent to the GTF for further treatment.

Most of the solids can be dissolved in nitric acid which, when separated from the undissolved solids, becomes the feed for the TRUEX process. The TRUEX process separates a low-volume TRU concentrate stream away from the large-volume LLW stream (Schulz et al. 1980).

The LLW stream is combined with the LLW supernatant for treatment in the GTF. The TRU concentrate stream is combined with the undissolved solids for treatment in the HWVP.

#### 3.5.3 Schedule

Pilot plant testing of the PFP waste treatment flowsheet is scheduled for FY 1994 with full-scale processing scheduled for FY 2000 (Augustine 1989).

### 3.6 PLANNED TREATMENT OF COMPLEXANT CONCENTRATE WASTE

#### 3.6.1 Definition of Complexant Concentrate Waste

Complexant concentrate waste results from concentration of wastes containing large amounts of organic complexing agents. These organic compounds were introduced to the waste during strontium recovery processing in B Plant.

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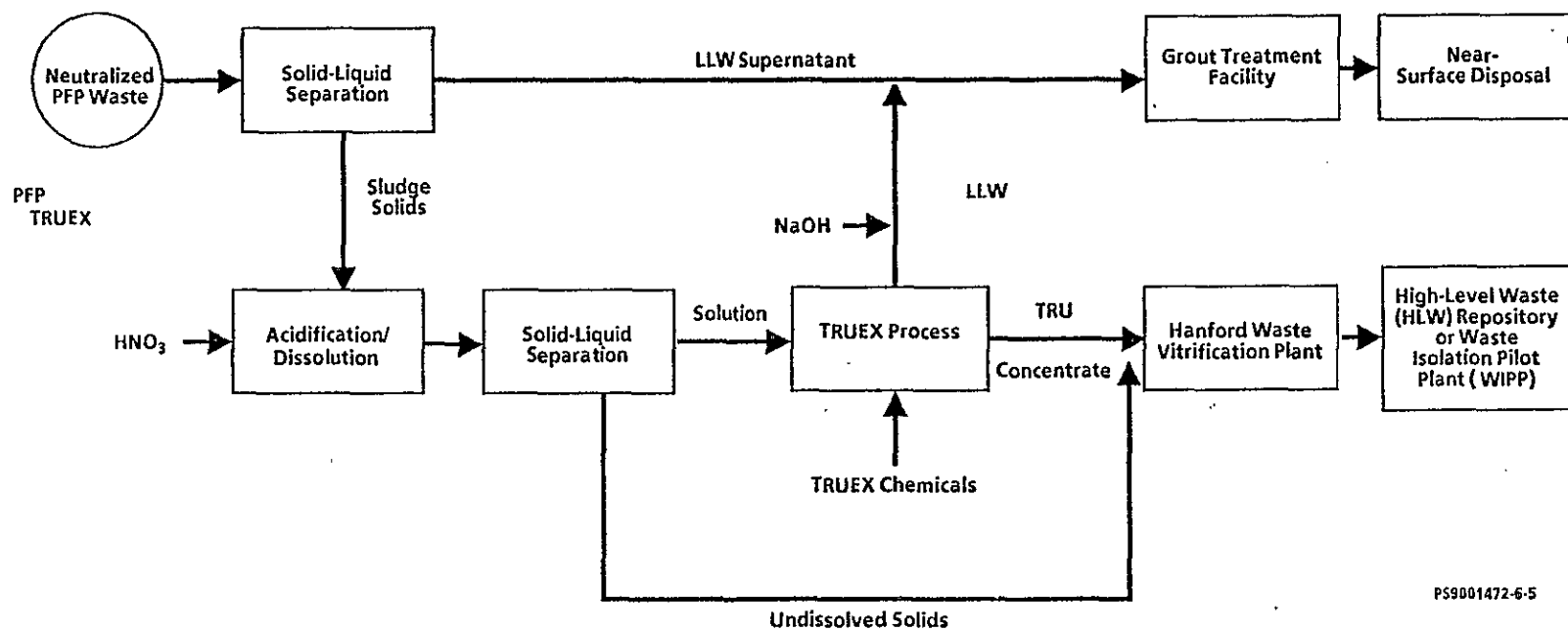


Figure 3-3. Plutonium Finishing Plant Waste Pretreatment and Disposal.

### 3.6.2 Planned Treatment of Complexant Concentrate Waste

The first step in the proposed treatment process is to acidify the CC waste stream to dissolve as many of the solids as possible (Figure 3-4) (Kurath 1985, 1986). The liquid is separated from the undissolved solids and used as feed to the TRUEX process.

The TRUEX process separates a low-volume TRU concentrate waste stream from a high-volume LLW stream containing organics and possibly cesium. The TRU concentrate stream is added to the undissolved solids and is treated in the HWVP.

The LLW stream containing organics and possibly cesium is treated for organic destruction and, if necessary, the cesium is removed (Lutton et al. 1980). The resulting LLW stream is sent to the GTF for conversion into grout.

### 3.6.3 Schedule

Pilot plant testing of the CC waste treatment process is scheduled for FY 1995. Full-scale processing of CC waste is scheduled for FY 2001 (Augustine 1989).

## 3.7 SUMMARY OF DOUBLE-SHELL TANK WASTE TREATMENT

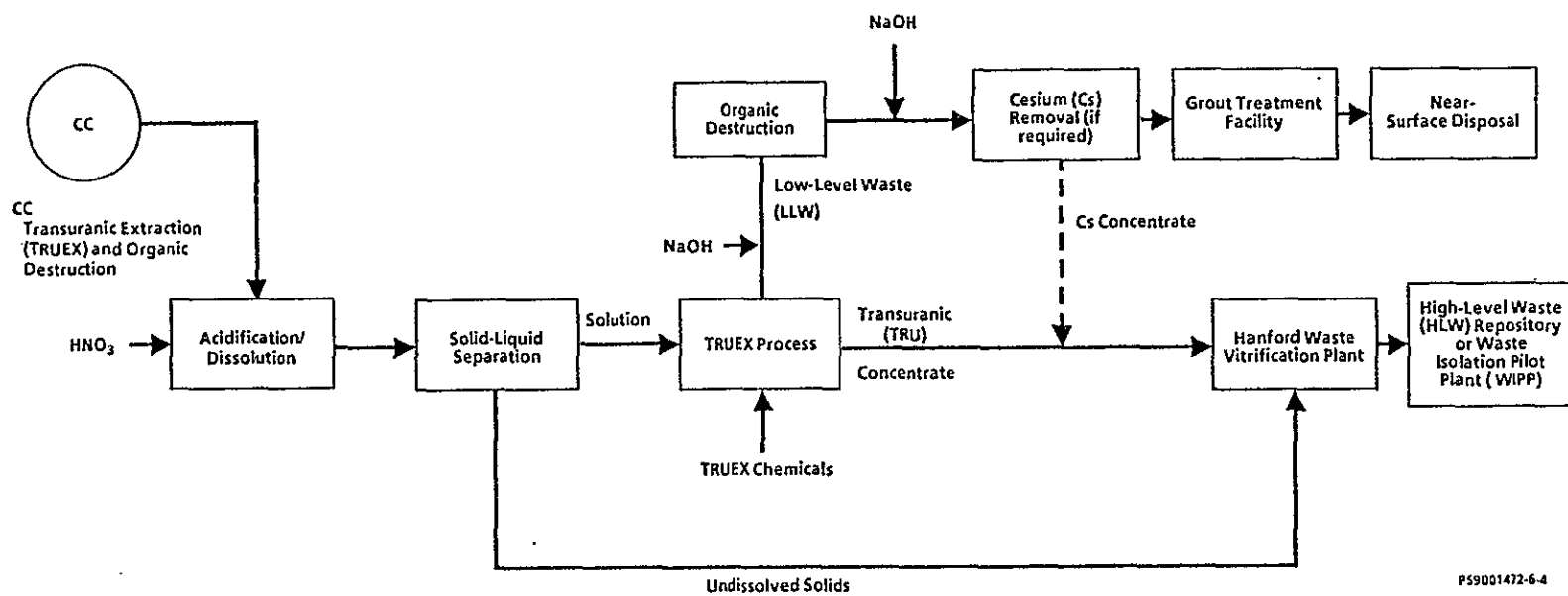
Studies have been performed to evaluate alternative processes and facilities for treatment of DST wastes before final disposal. A 1989 study confirmed the technical and economic incentives for partitioning the waste into a large, low-level fraction suitable for near-surface disposal, and a smaller fraction of TRU waste and/or HLW that must be immobilized by solidification in glass (Kupfer et al. 1989).

An evaluation of alternative facilities for performing waste treatment processes and optimum schedules for timely completion of the DST waste disposal mission was completed in 1990. The evaluation defined the existing baseline waste treatment plan for DST waste:

- Separate NCAW sludges from supernatant liquids and wash the sludge with water to remove soluble salts.
- Remove TRU waste components from acidified wastes using the TRUEX process. This technology is being developed for application with NCRW, PFP waste, and CC waste.
- Remove radiocesium from alkaline NCAW supernatant liquors.
- Destroy the complexant in CC waste to remove complexed TRU elements and provide a feed for near-surface disposal.



Figure 3-4. Complexant Concentrate Waste Pretreatment and Disposal.



Waste treatability tasks that are in progress or expected to be initiated in FY 1990 are described below. Documentation describing the results of these studies will be provided in future annual reports.

- Continue laboratory-scale tests to assess the application of the TRUEX process to remove TRU components from acidic solutions of actual NCRW, PFP waste, and CC waste.
- Perform laboratory-scale tests of organic complexant destruction methods. Methodologies under investigation include ozone-ultraviolet light, hydrogen peroxide-ultraviolet light, and supercritical water.
- Perform preliminary design of a pilot-scale facility for demonstrating the TRUEX process with actual DST wastes.
- Complete a siting study for the organic destruction pilot plant.
- Provide preliminary conceptual flowsheets for the TRUEX process for pretreatment of NCRW, PFP waste, and CC waste.
- Conduct screening experiments of potential solvent extraction processes for recovery of strontium-90 ( $^{90}\text{Sr}$ ) and cesium-137 ( $^{137}\text{Cs}$ ) from acidified wastes. This technology is being explored at the Argonne National Laboratory (ANL).
- Perform capacity tests and aging and cycling tests of candidate ion-exchange resins for removal of  $^{137}\text{Cs}$  from alkaline waste.

### 3.8 EXISTING TANK DOUBLE-SHELL TANK INVENTORY

The current waste volume inventory of the Hanford Site tank farms as of February 1990 is listed in Table 3-1. This information is available from the *Tank Farm Surveillance and Waste Status Summary Report for April 1990*, WHC-EP-0182-25 (Hanlon 1990). The volumes of both solids and liquids are in thousands of gallons. Several documents contain assessments of the existing and projected DST wastes stored at the Hanford Site:

1. DOE/EIS-0113, *Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington* (DOE 1987). This document addresses the existing DST wastes as of December 1987.
2. DOE/RW-0006, *Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics* (DOE 1989). This document addresses the existing DST wastes as of December 31, 1989.
3. SD-WM-TI-335, *Methods and Data for Use in Determining Source Terms for the Grout Disposal Program* (Richmond 1988). Table 4.1.3 of SD-WM-TI-335 addresses the accumulated volume and radioactivity, representative chemical composition, and representative radioactive composition of DSSF and DSS wastes stored at the Hanford Site as of 1988 and projected through FY 2020.

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Document SD-WM-TI-355 also characterized DSSF and DSS wastes for analysis of source terms for the Grout Disposal Program (Richmond 1988). Tanks 241-AN-103, 241-AN-106, and 241-AW-101 were sampled and the resulting information is presented for radioisotopes in Table 4.1.4, for inorganic chemical constituents in Table 4.1.5, and for organic chemical constituents in Table 4.1.6 of SD-WM-TI-355.

Table 3-1. Double-Shell Tank Inventory as of February 1990. (sheet 1 of 2)

Tank	Waste type <sup>a</sup>	Volume (kgal)			
		Liquid	DSS	Sludge	Saltcake
241-AN-101	NCPLX	235	0	0	--
241-AN-102	CC waste	1,020	0	89	0
241-AN-103	DSS	10	937	0	--
241-AN-104	DSSF	826	0	264	0
241-AN-105	DSSF	1,128	0	0	0
241-AN-106	CHFW	1,010	0	17	0
241-AN-107	CC waste	956	0	134	0
241-AP-101	NCPLX	1,065	0	0	0
241-AP-102	HFW <sup>b</sup>	134	0	0	0
241-AP-103	NCPLX	590	0	0	0
241-AP-104	HFW	22	0	0	0
241-AP-105	DSSF	828	0	0	0
241-AP-106	NCPLX <sup>c</sup>	1,138	0	0	0
241-AP-107	NCPLX	1,139	0	0	0
241-AP-108	NCPLX	78	0	0	0
241-AW-101	DSSF	1,038	0	84	0
241-AW-102	NCPLX	996	0	1	0
241-AW-103	NCPLX	319	0	363	0
241-AW-104	NCPLX	572	0	179	111
241-AW-105	NCPLX	635	0	297	0
241-AW-106	DSSF	295	0	198	85
241-AY-101	CPLX	841	0	83	0
241-AY-102	NCPLX	570	0	32	0
241-AZ-101	NCAW	924	0	37	0
241-AZ-102	NCAW	857	0	88	0
241-SY-101	CC waste	253	561	0	560
241-SY-102	NCPLX	503	0	71	0
241-SY-103	DSS	177	573	0	4
Total		18,159	2,071	1,937	760

<sup>a</sup>See sheet 2 of 2 for description.<sup>b</sup>The 18.6 in. of total waste in Tank 241-AP-102 is GTF excess waste.<sup>c</sup>The 77 in. of total waste in Tank 241-AP-106 is DSSF.

Table 3-1. Double-Shell Tank Inventory as of February 1990.  
(sheet 2 of 2)

Waste type abbreviation	Waste type	Description
NCAW	Aging waste	The high-level, first-cycle solvent extraction waste from the PUREX Plant.
CPLX	Complexant waste	Dilute waste material containing relatively high concentrations of chelating agents, such as EDTA and HEDTA, from B Plant waste fractionization operation.
CC waste	Complexant concentrate	The product of concentrating complexed waste.
HFV	Hanford Site facility waste	Waste that is generated in the 100-N, 300, and 400 Areas and refers to a waste source (as opposed to a waste type). This category includes two waste types: dilute phosphate and dilute, noncomplexed waste.
CHFW	Concentrated Hanford Site facility waste	The product of concentrating Hanford Site facility waste.
NCPLX	Noncomplexed	A general waste term applied to all Hanford Site liquors not identified as complexed.
DSS	Double-shell slurry	The product of concentrating DSSF past the sodium aluminate boundary to a solid-liquid matrix containing interstitial liquid. The interstitial liquid may not be drainable. For reporting purposes, DSS is considered a solid.
DSSF	Double-shell slurry feed	NCPLX waste that has been concentrated until the solution is near the sodium aluminate saturation boundary.

EDTA = ethylenediaminetetraacetic acid  
HEDTA = hydroxyethylenediaminetriacetic acid

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## 4.0 TREATMENT OF EXISTING SINGLE-SHELL WASTES

### 4.1 DESCRIPTION OF SINGLE-SHELL TANK WASTES

One hundred and forty-nine SSTs contain portions of HLW, TRU waste, and LLW produced during Hanford Site operations before 1980. The current waste inventory of the SST system as of February 1990 is given in Table 4-1 (Hanlon 1990). Interim stabilization efforts are currently under way to remove pumpable liquid from the SSTs leaving saltcake, sludge, and minimal interstitial liquid. This supports Tri-Party Agreement Interim Milestone M-05-09. The remaining SST contents form the basis for future treatment efforts.

### 4.2 TREATMENT OF SINGLE-SHELL TANK WASTES

Two treatment alternatives are being considered: in situ treatment and treatment after retrieval.

In situ treatment will involve a stabilization that will negate the retrieval of any SST waste.

The treatment-after-retrieval alternative has two goals: (1) minimize the fraction of waste fed to the HWVP while meeting current DST feed chemical concentration limits, and (2) maximize the fraction of nonradioactive chemical compounds routed to GTF while meeting the non-TRU ( $<100$  nCi/g),  $^{90}\text{Sr}$ , and  $^{137}\text{Cs}$  concentration requirements for the solidified grout. The processes for treatment of the retrieved SST waste are based on the processes and equipment being developed by the DST program: sludge washing, TRUEX, cesium ion exchange, and possibly complexant destruction. Treatment technologies specific to SST waste are being studied and funded by the Environmental Restoration (ER) Program.

### 4.3 STATUS OF SINGLE-SHELL TANK WASTE STUDIES

The following information provides the status of SST waste treatment activities completed and/or in progress. As noted, some of the development activities listed are being funded by and for the DST program.

#### 4.3.1 Destruction of Complexant Concentrate Waste

Research in the area of the destruction of CC waste is funded through the DST program and applies to the destruction of CC waste in the DSTs. However, the developing technology may have application to the variety of CC waste that may be in the SSTs (Winters 1981).

#### 4.3.2 Removal of Transuranic Components by the TRUEX Process

The research into the possibility of removing the TRU components of both SST and DST wastes is funded by the DST program for application to DST waste. The successful development of technology to remove the TRU waste content of the DST wastes may have direct application to treatment of SST waste.

#### 4.3.3 The Strontium Extraction Process

A new process is being developed for the extraction and recovery of  $^{90}\text{Sr}$  from acidic nuclear waste streams. It is called the SREX process for "strontium extraction."

The funding for this research effort is being phased out by the DST program and is being phased in by the SST program. The ANL Laboratory is performing the research to explore processes for the recovery of  $^{90}\text{Sr}$  and possibly  $^{137}\text{Cs}$  from acidic liquid HLW.

The new strontium extraction recovery process (based on a crown ether) has been demonstrated to be an effective and selective solvent extraction process. Continued research is necessary to determine its feasibility on a plant scale. Items to investigate include, but are not necessarily limited to, entrainment losses, radiation effects on the process solvent, and the extraction behavior of other fission products.

#### 4.4 SELECTION OF SINGLE-SHELL TANK WASTE FORM

A literature search is currently being conducted by the Westinghouse Hanford Company (Westinghouse Hanford) and the Pacific Northwest Laboratory (PNL). Throughout the course of this study Westinghouse Hanford and PNL will examine SST waste compatibility with vitrified and grouted waste forms. They will also perform a scoping study to examine in-tank treatment options for complexant concentrates.

#### 4.5 LABORATORY STUDIES

Westinghouse Hanford is currently conducting laboratory studies to develop acid leach data for the SST wastes. Acid washing capabilities as a unit operation in the treatment of SST waste will depend on developed data to determine its applicability, effectiveness, and possible equipment specification criteria.

Current acid leach data are based on laboratory use of hydrochloric acid. This type of acid is impractical for use in a plant-sized acid treatment process; therefore, acid leach data is being generated that will use acid types commensurate with plant-sized operations.

Westinghouse Hanford is conducting tests that use more appropriate acid types. These data will be useful in determining sizing requirements and required processing of SST waste.



#### 4.6 ENGINEERING STUDIES

A current-year study that uses a systems engineering approach to examine the closure of the SST waste system is under way (Klem et al. 1990). Sections of this study are dedicated to the treatment of both in situ and retrieved SST waste. Issues addressed will be treatment methods and facility requirements.

#### 4.7 BACKGROUND INFORMATION

A compilation of pertinent treatability-related reports and documents generated at the Hanford Site, as well as several reports from other sites that address relevant SST waste issues are included in Section 8.0. Abstracts for these references are also provided for additional information.

Table 4-1. Single-Shell Tank Inventory as of February 1990.  
(sheet 1 of 5)

Tank	Waste type <sup>a</sup>	Volume (kgal)		
		Supernatant liquid	Sludge <sup>b</sup>	Saltcake
241-T-101	NCPLX	30	103	0
241-T-102	NCPLX	13	19	0
241-T-103	NCPLX	3	23	0
241-T-104	NCPLX	3	442	0
241-T-105	--	23	98	0
241-T-106	NCPLX	2	19	0
241-T-107	NCPLX	9	171	0
241-T-108	NCPLX	0	44	0
241-T-109	--	0	58	0
241-T-110	NCPLX	3	376	0
241-T-111	NCPLX	2	456	0
241-T-112	NCPLX	7	60	0
241-T-201	--	1	28	0
241-T-202	--	0	21	0
241-T-203	--	0	35	0
241-T-204	--	0	38	0
241-A-101	DSSF	0	3	950
241-A-102	DSSF	4	15	22
241-A-103	DSSF	4	373	0
241-A-104	--	0	28	0
241-A-105	--	0	19	0
241-A-106	CHFW	0	125	0
241-AX-101	DSSF	0	3	745
241-AX-102	CC	3	7	29
241-AX-103	--	0	2	110
241-AX-104	--	0	7	0
241-B-101	--	0	113	0
241-B-102	NCPLX	4	18	10
241-B-103	NCPLX	0	59	0
241-B-104	NCPLX	1	301	69
241-B-105	--	0	40	266
241-B-106	NCPLX	1	116	0
241-B-107	NCPLX	1	164	0
241-B-108	NCPLX	0	94	0
241-B-109	NCPLX	0	127	0
241-B-110	NCPLX	1	244	0
241-B-111	NCPLX	1	236	0
241-B-112	NCPLX	3	30	0
241-B-201	NCPLX	1	28	0
241-B-202	NCPLX	1	26	0

Table 4-1. Single-Shell Tank Inventory as of February 1990.  
(sheet 2 of 5)

Tank	Waste type <sup>a</sup>	Volume (gal)		
		Supernatant liquid	Sludge <sup>b</sup>	Saltcake
241-B-203	NCPLX	1	49	0
241-B-204	NCPLX	1	49	0
241-BX-101	NCPLX	1	42	0
241-BX-102	--	0	96	0
241-BX-103	NCPLX	4	62	0
241-BX-104	NCPLX	3	96	0
241-BX-105	NCPLX	4	43	3
241-BX-106	NCPLX	15	31	0
241-BX-107	NCPLX	9	361	0
241-BX-108	--	0	26	0
241-BX-109	NCPLX	2	197	0
241-BX-110	NCPLX	2	189	7
241-BX-111	NCPLX	19	68	143
241-BX-112	NCPLX	13	157	0
241-BY-101	--	0	109	278
241-BY-102	NCPLX	15	0	417
241-BY-103	NCPLX	0	5	405
241-BY-104	--	0	40	366
241-BY-105	--	0	44	459
241-BY-106	NCPLX	0	95	547
241-BY-107	--	0	60	206
241-BY-108	NCPLX	0	154	74
241-BY-109	NCPLX	33	87	354
241-BY-110	NCPLX	0	103	295
241-BY-111	NCPLX	0	21	438
241-BY-112	NCPLX	0	5	286
241-C-101	--	0	88	0
241-C-102	NCPLX	3	424	0
241-C-103	NCPLX	25	175	0
241-C-104	CPLX	0	295	0
241-C-105	CPLX	0	150	0
241-C-106	CPLX	32	197	0
241-C-107	--	0	337	0
241-C-108	--	0	65	0
241-C-109	NCPLX	4	62	0
241-C-110	NCPLX	5	196	0
241-C-111	--	0	57	0
241-C-112	--	0	109	0
241-C-201	NCPLX	1	2	0

Table 4-1. Single-Shell Tank Inventory as of February 1990.  
(sheet 3 of 5)

Tank	Waste type <sup>a</sup>	Volume (kgal)		
		Supernatant liquid	Sludge <sup>b</sup>	Saltcake
241-C-202	--	0	1	0
241-C-203	NCPLX	0	5	0
241-C-204	NCPLX	0	3	0
241-S-101	DSSF	12	244	171
241-S-102	--	0	4	545
241-S-103	DSSF	17	10	221
241-S-104	NCPLX	1	293	0
241-S-105	--	0	2	454
241-S-106	--	0	32	511
241-S-107	DSSF	6	293	69
241-S-108	--	0	4	600
241-S-109	--	0	13	555
241-S-110	--	0	131	561
241-S-111	NCPLX	10	139	447
241-S-112	--	0	6	631
241-SX-101	NCPLX	1	112	343
241-SX-102	--	0	117	426
241-SX-103	DSSF	32	112	523
241-SX-104	DSSF	0	136	478
241-SX-105	--	0	73	610
241-SX-106	NCPLX	61	12	465
241-SX-107	--	0	104	0
241-SX-108	--	0	115	0
241-SX-109	--	0	250	0
241-SX-110	--	0	62	0
241-SX-111	--	0	125	0
241-SX-112	--	0	92	0
241-SX-113	--	0	26	0
241-SX-114	--	0	181	0
241-SX-115	--	0	12	0
241-TX-101	NCPLX	3	84	0
241-TX-102	--	0	0	113
241-TX-103	NCPLX	0	157	0
241-TX-104	CPLX	1	0	64
241-TX-105	--	0	0	609
241-TX-106	--	0	0	453
241-TX-107	NCPLX	1	0	35
241-TX-108	NCPLX	0	0	134
241-TX-109	--	0	0	384

Table 4-1. Single-Shell Tank Inventory as of February 1990.  
(sheet 4 of 5)

Tank	Waste type <sup>a</sup>	Volume (kgal)		
		Supernatant liquid	Sludge <sup>b</sup>	Saltcake
241-TX-110	--	0	0	462
241-TX-111	--	0	0	370
241-TX-112	--	0	0	649
241-TX-113	--	0	0	607
241-TX-114	--	0	0	535
241-TX-115	--	0	0	640
241-TX-116	--	0	0	631
241-TX-117	--	0	0	626
241-TX-118	--	0	0	347
241-TY-101	--	0	118	0
241-TY-102	NCPLX	0	0	64
241-TY-103	--	0	162	0
241-TY-104	NCPLX	0	43	0
241-TY-105	--	0	231	0
241-TY-106	--	0	17	0
241-U-101	NCPLX	3	22	0
241-U-102	NCPLX	18	43	313
241-U-103	DSSF	13	32	423
241-U-104	NCPLX	0	122	0
241-U-105	CC	37	32	349
241-U-106	CC	15	26	185
241-U-107	DSSF	31	15	360
241-U-108	NCPLX	24	29	415
241-U-109	NCPLX	19	48	396
241-U-110	--	0	186	0
241-U-111	DSSF	0	26	303
241-U-112	NCPLX	4	45	0
241-U-201	NCPLX	1	4	0
241-U-202	NCPLX	1	4	0
241-U-203	NCPLX	1	2	0
241-U-204	NCPLX	1	2	0

<sup>a</sup>See sheet 5 of 5 for description.<sup>b</sup>Includes interstitial liquid.

Table 4-1. Single-Shell Tank Inventory as of February 1990.  
(sheet 5 of 5)

Waste type abbreviation	Waste type	Description
CPLX	Complexant waste	Dilute waste material containing relatively high concentrations of chelating agents, such as EDTA and HEDTA, from B Plant waste fractionization operation.
CC waste	Complexant concentrate	The product of concentrating complexed waste.
CHFW	Concentrated Hanford Site facility waste	The product of concentrating Hanford Site facility waste.
NCPLX	Noncomplexed	A general waste term applied to all Hanford Site liquors not identified as complexed.
DSSF	Double-shell slurry feed	NCPLX waste that has been concentrated until the solution is near the sodium aluminate saturation boundary.

## 5.0 SELECTION OF GROUT FOR DISPOSAL OF CERTAIN HANFORD SITE LOW-LEVEL LIQUID WASTES

Cement-based grouts are extensively used in the United States and elsewhere as a vehicle for immobilization and near-surface disposal of solid and liquid LLWs. Table 5-1 provides a brief chronological listing of important milestones in the 30-yr history of United States experience in grout disposal of certain liquid nuclear wastes at the DOE-operated Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee; Savannah River Laboratory (SRL), Aiken, South Carolina; and Hanford Site locations.

Formal selection of cementitious grout for disposal of selected Hanford Site liquid wastes in near-surface vaults was made in late 1983 with the publication of the first edition of the *Hanford Waste Management Plan* (DOE-RL 1983). Selection of a grout waste form for use at the Hanford Site was strongly influenced by the generally favorable previous ORNL Site grout hydrofracture disposal experience and by the SRL Site evaluation and selection of a grout waste form for the disposal of certain low-level radioactive aqueous salt solutions. More significantly, independent, detailed, and comprehensive evaluations performed by highly qualified Hanford Site scientists and engineers in 1980 showed grout to be preferred over other known forms for immobilization and bulk disposal of Hanford Site liquid LLW (RHO 1980a).

Table 5-2 summarizes relevant results of the 1980 evaluation of grout and several other candidate waste forms for immobilization and disposal of wastes stored in SSTs and DSTs at the Hanford Site.

Table 5-1. Chronology for the Selection of the Grout Waste Form for Immobilization of Hanford Site Liquid Low-Level Waste: Important Milestones in the United States.

Event	Description	Time
1	Disposal of United States commercial nuclear reactor LLW in grout	1960 to present
2	Operation of ORNL site hydrofracture process and facilities	1966 to 1979; 1982 to 1984
3	Brookhaven National Laboratory studies of concrete and concrete waste forms	1970s
4	First recognition of need to select an Immobile form for the disposal of Hanford Site liquid LLW (ERDA 1977)	September 1977
5	SRL Site evaluation of grout and other forms for immobilization of HLWs	1979
6	Hanford Site evaluation of grout and other forms for immobilization of HLW (Schulz et al. 1980)	September 1980
7	SRL Site decision to dispose of decontaminated liquid LLW in grout waste form	February 1982
8	Formal Hanford Site decision to dispose of liquid LLW in grout form (RHO 1983)	December 1983
9	Preparation of Integrated Grout Management Plan and further review of suitability of grout as waste disposal form (RHO 1986)	November 1986
10	First plant-scale grout disposal campaign at the Hanford Site	August 1988



Table 5-2. The 1980 Relative Ranking of Selected Waste Forms for Immobilization of Hanford Site Blended Wastes.<sup>a, b</sup>

Rank	Waste form	Numerical scores		
		Overall	Form <sup>c</sup>	Process <sup>d</sup>
1	Borosilicate glass-monoliths	65	33	32
2	Grout	66	23	43
3	Bitumen	62	21	41
4	Aqueous Silicate	60	17	43

<sup>a</sup>Data abstracted from RHO (1980b).

<sup>b</sup>For blended liquid and solid wastes from Hanford Site tanks.

<sup>c</sup>Included evaluation of the following attributes:

- Status of development
- Scale-up potential
- Quality assurance
- Process safety
- Simplicity
- Remoteability
- Rework capacity.

<sup>d</sup>Included evaluation of the following attributes:

- Waste loading
- Leachability
- Thermal stability
- Repository environment and waste form interaction
- Radiation stability
- Mechanical stability
- Fire resistance.

Grout along with bitumen (asphalt) and aqueous silicate\* waste forms were considered in the 1980 evaluation to be the leading candidate waste forms and processes for bulk disposal (as opposed to individual canisters) of Hanford Site SST wastes. Ranking and evaluation data listed in Table 4-2 show that a grout waste was judged to be superior to either bitumen or aqueous silicate forms for immobilization of blended (liquid plus solids) Hanford Site SST wastes.\*\*

Large-scale production of the grout, aqueous silicate, and bitumen forms were all judged to be comparably convenient. Properties of the aqueous silicate form, especially leachability, are inferior to those of a grout form, but the bitumen and grout forms have similar satisfactory properties. The bitumen form is 10 to 100 times more resistant than the grout form to water leaching of incorporated inert and radioactive components. The bitumen form is quite susceptible to damage by fire, and there is no significant United States experience in large-scale operation of a facility to incorporate waste into bitumen.

The choice of a grout waste form for immobilization and surface disposal of Hanford Site radioactive liquid LLW was further examined in 1986 as part of preparation of the Integrated Management Plan (RHO-WM-PL-12P) (RHO 1986). In the 1986 review, cementitious grout was compared to thermoplastic resins and asphalt (e.g., bitumen) and organic polymers (urea-formaldehyde and vinyl-ester styrene) forms. As in the 1980 evaluation, grout was found preferable to the other candidate waste forms for plant-scale manufacture and disposal of Hanford Site radioactive LLW liquors. Major advantages and disadvantages of the cementitious grouting process identified in the 1986 study are shown in Table 3 of RHO-WM-PL-12P (RHO 1986).

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\*The aqueous silicate waste form is produced by adding a naturally occurring clay to alkaline solutions or slurries; the mineral cancrinite is one of the principal constituents of the final aqueous silicate form.

\*\*Indeed, a grout form was judged overall to be comparable to the reference borosilicate glass monolithic form; the leachability and other properties of grout were found to be inferior to those of glass, but a grout form was considered to be easier and simpler to prepare than a high-temperature glass.

Table 5-3. Major Advantages and Disadvantages Associated with the Use of Cementitious Grout as a Fixation Medium.

Advantages	Disadvantages
<ul style="list-style-type: none"> <li>• Compatible with many types of waste.</li> <li>• Minimal free-standing liquid when properly engineered.</li> <li>• Moderate-to-good resistance from leaching.</li> <li>• Relatively inexpensive, locally available materials and process equipment.</li> <li>• Process experience readily available.</li> <li>• Pumpability as a slurry; therefore, high productivity and low exposure potential.</li> <li>• No process side streams are generated.</li> <li>• Large-scale monolith casting is easily facilitated, reducing the surface area-to-volume ratio.</li> <li>• Processing occurs at ambient temperature, minimizing volatilization.</li> <li>• Waste form is easily tailored to produce desired physical or chemical retention properties.</li> <li>• Resistant to radioactive degradation.</li> <li>• Waste immobilized by chemical combination with cement constituents to form partially hydrated compounds, containment in pore structure of grout matrix, and by mechanical bindings of solid particles by grout matrix.</li> </ul>	<ul style="list-style-type: none"> <li>• High levels of some metals, salts, or organics can slow or prevent curing.</li> <li>• Low volumetric efficiency.</li> <li>• Acidic waste must be neutralized.</li> <li>• Powdery nature of dry-mix may cause process variability.</li> <li>• Depending upon disposal scenario, long cure times may be required to achieve the desired physical character.</li> <li>• Low-to-moderate compressive strength compared to other options (well above regulatory criteria).</li> </ul>

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## 6.0 EVALUATION AND SELECTION OF BOROSILICATE GLASS

The following is a summary of the information presented in a white paper dealing with the suitability of borosilicate glass as the waste form of choice for the disposal of HLW. The complete paper is provided in Appendix B.

Borosilicate glass is the waste form of choice for the disposal of HLW worldwide. Thirteen production plants are currently in operation or under construction for the conversion of liquid HLW to borosilicate glass. Eleven of these plants are in foreign countries; two are located in the United States. One of the two United States plants is located at the SRL and the other at the West Valley Site in West Valley, New York. The HWVP will be the third United States plant, and fourteenth worldwide to be used for the conversion of nuclear HLW to borosilicate glass.

The selection of borosilicate glass as the waste form of choice at the three United States sites was the result of a thorough technical review of the alternatives at each of the sites and public comment through the National Environmental Policy Administration (NEPA) Act process at each of the sites (DOE 1987). There has been an important evolution in disposal regulations and applicable tests for waste forms since the decisions were made. A review of borosilicate glass data as they apply to the new waste form tests and evolving regulations shows that borosilicate glass remains a fully satisfactory waste form for HLW immobilization in the HWVP (Schulz et al. 1980).

Several alternative waste forms evaluated for use in the United States:

1. Synroc, a hot-pressed crystalline waste form, using naturally occurring minerals known to survive a long time in nature
2. Tailored ceramics, a predominantly crystalline waste form whose crystalline phases are chosen on material science rather than geological considerations
3. Porous glass matrix (high-silica glass), a somewhat heterogeneous waste form, produced by sintering and intimate mixture of calcined waste and porous glass powder at 1200 °C
4. Concrete (FUETAP), a heterogeneous crystalline waste form, consisting of wastewater, cement and other solid powder additives mixed and cured under elevated temperatures (100 to 250 °C) and pressures (1000 lb/in<sup>2</sup>) to form a monolith
5. Coated Sol-Gel Particles, a heterogeneous waste form consisting of small (<1 mm), predominantly crystalline, waste-containing particles coated with three layers (low-density SiC, high-density SiC and high-density pyrolytic C) and embedded in an inorganic binder

6. Glass marbles in a lead matrix, a heterogeneous waste consisting of borosilicate glass marble (1.3 cm in diameter) embedded in a continuous matrix of lead-tin alloy.

The borosilicate glass waste form was selected because it readily accommodates fluctuations in waste composition and remote operation of the processing equipment is a well-proven technology. Synroc, or one of the many other alternatives to borosilicate glass that have been proposed, could emerge as a potentially feasible second-generation waste form for use in the future, but numerous questions remain to be answered before their adoption. A large nonradioactive Synroc pilot plant is in operation, but the Synroc process has not yet been tested with radioactive materials, except for a few laboratory-scale experiments. In the interval, many studies have shown the suitability of borosilicate glass and borosilicate glass is the first-generation waste form for HLW throughout the world.

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- Schulz, W. W., M. M. Beary, S. A. Gallagher, B. A. Higley, R. G. Johnston, F. M. Jungfleisch, M. J. Kupfer, R. A. Palmer, R. A. Watrous, and G. A. Wolf, 1980 (September), *Preliminary Evaluation of Alternative Forms for Immobilization of Hanford High-Level Defense Wastes*, RHO-ST-32, Rockwell Hanford Operations, Richland, Washington.
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Winters, W. I., 1981 (June), *Effect of pH on the Destruction of Complexants with Ozone in Hanford Nuclear Waste*, RHO-SA-203, Rockwell Hanford Operations, Richland, Washington.

Wong, J. J., 1989 (October), *244-AR Conceptual Flowsheet for Processing of NCAW*, WHC-SE-WM-TI-396, Revision 0, Westinghouse Hanford Operations, Richland, Washington.

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ANCI, 1975 (December), *High Level Waste Management Technology Development Program*, 1060-19, Prepared by Applied Nucleonics Company, Inc., Santa Monica, California, for Atlantic Richfield Hanford Company, Richland, Washington.

In January 1975 a program was established to evaluate technology for retrieval of solid and liquid wastes presently stored on the Hanford Reservation. The original program outline involved three tasks: 1) waste retrieval, 2) volume reduction and waste immobilization, and 3) equipment decommissioning. In March 1975 the scope of work was expanded to include a fourth task, development of concepts for an engineered storage facility.

"On the basis of thorough evaluation of existing information on the high level waste management alternatives, technologies, and criteria and on the basis of independent engineering analysis, the following recommendations were made:

- In retrieval operations, the reference design should be based on the mechanical in-tank material handling.
- In waste stabilization operations, the reference design should be based on the silicate melt processing alternative.
- In engineered storage operations, the reference design should be based on the modular bin concept.
- Finally, in equipment decommissioning operations, additional research and development should be conducted prior to the establishment of a reference design.

Augustine, C. A., 1989 (January), *Double-Shell Tank Waste Disposal Integration Plan*, WHC-EP-0229, Revision 1, Westinghouse Hanford Company, Richland, Washington.

Bates, S. O., G. F. Piepel, and J. W. Johnston, 1989 (May), *Leach Testing of Simulated Hanford Waste Vitrification Plant Reference Glass HW-39*, PNL-6884, Prepared by Pacific Northwest Laboratory for Westinghouse Hanford Company, Richland, Washington.

The document summarizes the work performed to investigate the viability of a leach testing methodology for the HWVP and provide glass dissolution data for HWVP model determination and validation. Leach tests up to one year in duration were conducted on the reference glass HW-39-1. Some changes are recommended to the leach test methodology.

Buckley, L. L., and J. D. Kaser, 1983 (August), *Costs of Alternatives for the Disposal of Future PUREX High-Level Waste and Existing Waste in Double-Shell Tanks*, SD-WM-ES-019, Revision 0, Rockwell Hanford Operations, Richland, Washington.

The costs of selected alternatives for waste disposal were estimated. The cost effects of direct waste neutralization and byproduct removal were estimated. Costs for major waste processing and disposal alternatives were identified. The conclusions drawn are:

- All viable alternatives require operation of B Plant.
- Removal of cesium is cost-effective from a waste management standpoint.
- The cost of purification and encapsulation of cesium for beneficial use is comparatively small.
- Removal of strontium is not cost effective from a waste management standpoint.
- Purification and encapsulation costs for strontium are substantial.
- Relative to the Savannah River approach, use of B Plant to remove cesium allows utilization of existing Hanford capabilities and experience with no apparent cost penalty.

DeFigh-Price, C., and B. A. Higley, 1985 (June), *Project Engineering Bases for Treatment of Double-Shell Tank Wastes: Process and Facilities Options*, SD-WM-ES-065, Revision 0, Rockwell Hanford Operations, Richland, Washington.

This document contains cost estimates of alternative waste management plans and schedules prepared by Science Applications, Inc., Richland, for Rockwell Hanford Operations as 'Project Engineering Bases for Treatment of Double-Shell Tank Wastes' Process flowsheets on which the cost estimates are based, are presented in SD-WM-ES-025, 'Preliminary Process Flowsheets for Treatment of Hanford Defense Liquid Wastes.' These reports are summarized in SD-WM-ES-023, 'Evaluation of Process and Facility Options for Treatment of Double-Shell Tank Wastes.'

DOE, 1981 (July), *The Evaluation and Review of Alternative Waste Forms for Immobilization of High-Level Radioactive Wastes*, DOE/TIC-11472, U.S. Department of Energy, Washington, D.C.

The document presents the relative merits and potential of eight alternative waste forms for disposal of high-level radioactive waste. The eight waste forms were selected from 15 previously evaluated. A Peer Review Panel composed of eight scientists and engineers representing independent, non-DOE laboratories from industry, government, and universities and disciplines of material science, ceramics, glass, metallurgy, and geology did the review. The waste forms were ranked as follows: borosilicate glass, SYNROC, porous glass matrix, tailored ceramics, pyrolytic C and SiC coated particles, FUETAP concrete, metal matrices, and plasma spray coatings.

DOE, 1981 (August), *The Evaluation and Selection of Candidate High-Level Waste Forms*, DOE/TIC-11611, U.S. Department of Energy, Savannah River Operations Office, Aiken, South Carolina.

Seven candidate waste forms developed under the direction of the DOE's National High-Level Waste Technology Program, were evaluated as potential media for the immobilization and geologic disposal of high-level nuclear wastes. The evaluation combined preliminary waste form evaluations conducted at the DOE defense waste sites and independent laboratories, peer review assessments, a product performance evaluation, and a processability analysis. Based on the combined results of these four inputs, two of the seven forms, borosilicate glass and a titanate based ceramic, SYNROC, were selected as the reference and alternative forms for continued development and evaluation in the National HLW Program. Both the glass and ceramic forms were depicted as viable candidates for use at each of the DOE defense waste-sites. This report describes the waste form screening process, and discusses each of the four major input considered in the selection of the two forms.

DOE, 1982 (July), *Environmental Assessment, Waste Form Selection for SRP High-Level Waste*, DOE/EA-0179, U.S. Department of Energy, Washington, D.C.

This document presents borosilicate glass as the proposed waste form for disposal of SRP HLW, and crystalline ceramic as the leading alternative. A description of the properties, processing requirements, and development requirements for the proposed and alternate waste forms is provided. An assessment of the environmental consequences of the use of these two waste forms is presented. The document also lists 17 candidate waste forms that were considered for geologic disposal and describes the screening process by which borosilicate glass and crystalline ceramic were selected for further development.

DOE, 1987 (December), *Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington*, DOE/EIS-0113, 5 volumes, U.S. Department of Energy, Washington, D.C.

The purpose of this Environmental Impact Statement (EIS) is to provide environmental input into the selection and implementation of final disposal actions for high-level, transuranic and tank wastes located at Hanford Site, Richland, Washington, and into the construction, operation and decommissioning of waste treatment facilities that may be required in implementing waste disposal alternatives. Specifically evaluated are a Hanford Waste Vitrification Plant, Transportable Grout Facility, and a Waste Receiving and Packaging Facility. Also an evaluation is presented to assist in determining whether any additional action should be taken in terms of long-term environmental protection for waste that was disposed of at Hanford prior to 1970 as low-level waste (before the transuranic waste category was established by the Atomic Energy Commission but which might fall into that category if generated today).

The following alternatives are considered in this EIS: 1) in-place stabilization and disposal, where waste is left in place but is isolated by protective and natural barriers; 2) geologic disposal, where most of the waste (by activity and to the extent practicable is exhumed, treated, segregated, packaged and disposed of in a deep geologic repository; waste classified as high-level would be disposed of in a commercial

repository developed pursuant to the Nuclear Waste Policy Act; transuranic waste would be disposed of in the Waste Isolation Pilot Plant near Carlsbad, New Mexico; 3) a reference alternative, where some classes of waste are disposed of in geologic repositories and other classes of waste are disposed of by in-place stabilization and disposal; 4) the preferred alternative, in which double-shell tank wastes, strontium and cesium capsules, and retrievably stored TRU wastes are disposed of according to the reference alternative, and in which double-shell tank wastes, strontium and cesium capsules, and retrievably stored TRU wastes are disposed of according to the reference alternative, and in which decisions are deferred on disposal of single-shell tank wastes and on further remedial action for TRU-contaminated soil sites and pre-1970 buried suspect TRU-contaminated solid wastes (except the 618-11 site) until additional information is obtained on waste characterization, retrieval methods, and performance of new-surface disposal systems; and 5) a no disposal action alternative (continued storage).

DOE, 1989 (November), *Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Revision 5, Prepared by Oak Ridge National Laboratory, Oak Ridge, Tennessee, for U.S. Department of Energy, Headquarters, Washington, D.C.

DOE-RL, 1983, *Hanford Waste Management Plan*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

Dosch, R. G., 1978 (June), *The Use of Titanates in Decontamination of Defense Waste*, SAND78-0710, Prepared by Sandia Laboratories, Albuquerque, New Mexico, for Rockwell Hanford Operations, Richland, Washington.

Sodium titanate, an inorganic ion exchange material, has been evaluated for use in a process to remove strontium from Defense Waste or other sodium, caustic solutions. Distribution coefficients on the order of  $10^5$  were observed at sub part per million concentration of Sr, and the effects of other cation impurities and complexants in the waste were investigated.

The preparation and general chemical properties of the exchange material are discussed. This information was used in developing a commercial source which has since supplied a 200 kg batch of the material for evaluation.

In column ion exchange experiments with  $^{85}\text{Sr}$ -doped simulated waste, decontamination factors of 500 or greater were observed in the first 2000 to 3500 bed volumes of effluent, depending on the impurities in the simulant. A -40 to +130 mesh range of sodium titanate powder was used as the baseline material, but a study to produce alternate forms of the titanate was carried in parallel. This has resulted in two materials which appear promising with respect to both simplification of handling and chemical properties. One of the materials is an agglomerated form of the titanate formed by extrusion pelletizing using water as a binder, and the second is a macroreticular organic anion resin which was loaded with 30 to 40% (by weight) of sodium titanate. The results of initial testing of these materials are discussed.

Dunson, Jr., J. B., A. M. Eisenberg, R. L. Schuyler, III, H. G. Haight, Jr., V. E. Mello, T. H. Gould, Jr., J. L. Butler, and J. B. Pickett, 1982 (March), *Assessment of Processes, Facilities, and Costs for Alternative Solid Forms for Immobilization of SRP Defense Waste*, DP-1625, E. I. duPont, de Nemours & Company, Aiken, South Carolina.

The document presents a quantitative merit evaluation of the relative difficulty of remote processing of SRP HLW for seven waste forms. The borosilicate glass process is rated as the simplest. The other processes evaluated in order of increasing complexity were: FUETAP concrete, glass marbles in a lead matrix, high-silica glass, crystalline ceramic, and coated ceramic particles. Cost appraisals are summarized for the borosilicate glass, high-silica glass, and ceramic waste form processing facilities.

Ecology, EPA, and DOE, 1989 (May), *Hanford Federal Facility Agreement and Consent Order*, Washington State Department of Ecology, Olympia, Washington; U.S. Environmental Protection Agency, Region 10, Seattle, Washington; and U.S. Department of Energy, Richland, Washington.

ERDA, 1975 (December), *Final Environmental Statement, Waste Management Operations, Hanford Reservation, Richland, Washington*, ERDA-1538, 2 volumes, U.S. Environmental Research and Development Administration, Richland, Washington.

This Final Environmental Statement has been prepared toward compliance with the National Environmental Policy Act to assess the environmental impact of continuing the ERDA's waste management operations at the Hanford Reservation in Benton County, Washington.

Alternatives to current waste management operations are presented for radioactive and non-radioactive liquid, solid, and gaseous waste. Alternatives to the current waste management operations discussed include both the additional treatment of waste streams, discontinuance of solidification to salt cake and the reduction of waste generation by curtailment of operations at the site. Ceasing waste management operations is not considered due to the continuing need to manage existing radioactive waste at Hanford.

ERDA, 1977 (September), *Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Hanford Reservations, Richland, Washington*, ERDA 77-44, U.S. Energy Research and Development Administration, Washington, D.C.

The objective of this document is to provide information on alternatives that are being considered for the long-term management of defense high-level radioactive waste stored at Hanford in underground tanks and in stainless steel-lined concrete basins. For purposes of basic programmatic decisions making, four major alternatives based on disposal location are considered:

- existing waste tanks
- onsite engineered surface facilities
- onsite geologic repository
- offsite geologic repository.

The four major disposal alternatives are expanded into 27 alternative plans by considering:

- Variations in the final form of the high-level fraction (with radionuclide removal) to include glass, concrete, and powder.
- Variations in the final form of the dehydrated waste product to include glass, calcined clay, and powder.
- Variations in the treatment and handling of encapsulated waste to include packaging of capsules in canisters and conversion of the strontium fluoride and cesium chloride to glass.

A description of the technology, a preliminary risk assessment, and preliminary cost estimates for each of these 27 plans are presented.

ERDA, 1977 (May), *Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Savannah River Plant, Aiken, South Carolina*, ERDA 77-42/1, 2 volumes, U.S. Energy Research Administration, Richland, Washington.

This document was prepared to provide other Government agencies and the public information on possible alternatives which will be considered for the long-term management of Savannah River Plant (SRP) high-level nuclear waste. It describes a number of alternative plans for long-term management or disposal of the high-level nuclear wastes now stored in tanks at the Savannah River Plant near Aiken, South Carolina.

The Savannah River Plant operations produce high-level radioactive waste in the chemical processing of fuel and target elements after irradiation in the SRP nuclear reactors. This waste is stored as an alkaline liquid with a precipitated sludge until the decay heating has abated appreciable. The supernatant liquid is then converted to salt cake to reduce volume and mobility.

The purpose of the site-specific document is to describe the different alternatives along with their probable relative costs, risks, and uncertainties. A secondary purpose is to raise the issue of methodology for decision making in nuclear waste management. The document does not attempt to arrive at any recommendations.

Implementation costs and risk costs are calculated in the text for 23 alternative plans for long-range management and isolation of the SRP high-level radioactive waste. For purposes of basic programmatic decision making, these 23 plans can be grouped into four main classes (Figure II-1):

1. Convert the waste to a highly leach-resistant form, such as canned glass cylinders, and ship it offsite to a Federal repository.
2. Convert the waste to a highly leach-resistant form, and store the waste in an engineered surface facility at SRP.
3. Reconstitute the waste to a slurry, and dispose of it in a bedrock cavern under the SRP site.



4. Continue storage in tanks with the waste as salt cake and sludge."

Freeberg, R. D., 1989 "Tri-Party Agreement Milestone M-04-01" (External Letter 8905292 to P. T. Day, U.S. Environmental Protection Agency, Region 10, and R. G. Stanley, Washington State Department of Ecology, December 21, 1989), U.S. Department of Energy, Richland Operations Office, Richland, Washington.

Hammitt, A. P., and W. W. Schulz, 1978 (September), *Hot Cell Facility and Equipment for Test of the Hanford Radionuclide Removal Process*, RHO-SA-52, Rockwell Hanford Operations, Richland, Washington.

Bench-scale tests of ion exchange, precipitation, and other separation processes for removal of  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , actinides, and various other radionuclides from the water-soluble portion of the Hanford Defense Wastes have been successfully completed. This paper describes the hot cell and associated equipment to be used in further, scaled-up development and demonstration of the Hanford Radionuclide Removal process.

Hanlon, B. M., 1990 (June), *Tank Farm Surveillance and Waste Status Summary Report for April 1990*, WHC-EP-0182-25, Westinghouse Hanford Company, Richland, Washington.

Hannum, W. H., 1983 (January), *Analysis of the Terminal Waste Form Selection for the West Valley Demonstration Project*, DOE/NE/44139-T3, Prepared by West Valley Nuclear Services Company, West Valley, New York, for the U.S. Department of Energy, Washington D.C.

This document summarizes the environmental considerations associated with the selection of borosilicate glass as the waste form for the disposal of approximately 560,000 gallons of liquid HLW stored at West Valley. Product performance criteria discussed include leach resistance, thermal stability, mechanical stability, radiation stability, mechanical strength, impact resistance, fire resistance, and waste loading.

Higley, B. A., 1984 (April), *Preliminary Process Flowsheets for Treatment of Hanford Defense Liquid Wastes*, SD-WM-ES-025, Revision 0, Rockwell Hanford Operations, Richland, Washington.

This document compiles the preliminary process flowsheets which were prepared for use in estimating the costs of alternative waste management schedules and plans. Details of the cost estimates are presented in SAI-84-3013 "Project Engineering Bases for Treatment of Double Shell Tank Wastes: Process and Facility Options" while the "Evaluation of Process and Facility Options for Treatment of Double Shell Tank Wastes" is summarized in SD-WM-ES-023. These flowsheets account for the principal operations and capabilities required to pretreat and dispose of the waste as glass and grout. Eight flowsheets have been developed which describe the pretreatment, vitrification and transportable grout process.

They are:

- Waste removal and transfer
- Complexant destruction by ozonization
- Solids removal and washing
- Cesium removal by ion exchange
- Transuranic contaminant separation from cladding removal waste
- Low-level waste concentration
- Vitrification
- Low-level waste disposal by grout.

Higley, B. A., 1988 (January), *Impact of Alternative Single-Shell Tank Waste Retrieval and Pretreatment Scenarios on the Hanford Waste Vitrification Plant*, SD-WM-TA-014, Revision 0, Westinghouse Hanford Company, Richland, Washington.

This study examines the practicality of using the Hanford Waste Vitrification Plant (HWVP) to vitrify retrieved single-shell tank (SST) wastes. Four waste pretreatment alternatives for the retrieval of SST waste are considered. Three of the alternatives assume application of TRUEX process technology. The current HWVP design will allow installation of a 100 kg/hr glass melter without major modifications; this melter would be installed in the event that SST retrieval is required.

It is concluded that the HWVP as currently designed to accommodate a 100 kg/hr glass melter, is adequate to vitrify waste from a SST retrieval mission when TRUEX process technology is used in the waste pretreatment process. It is estimated that the use of TRUEX process technology could save 5,100 to 9,000 million dollars, depending on variations in the pretreatment process, relative to a base case in which washed sludge is vitrified.

Higley, B. A., and W. W. Schulz, 1988 (August), *Evaluation of Selected Alternatives for Processing Retrieved Hanford Single-Shell Tank Wastes*, WHC-EP-0191 DRAFT, Westinghouse Hanford Company, Richland, Washington.

Costs and various other impacts and features associated with the retrieval, processing, and immobilization (grout and/or vitrification) of various amounts of single-shell tank (SST) wastes were evaluated. Three different waste retrieval scenarios were considered: retrieval of wastes from 149, 75 and 12 SSTs. For each retrieval scenario, the effect of two processing treatments [simple sludge washing and sludge washing coupled with the Transuranic Extraction (TRUEX) process] on the final amount of disposed waste and on overall disposal costs was determined.

Cost savings from sludge washing coupled with the TRUEX process, when compared to simple sludge washing, range from 700 million to about 9 billion dollars depending on the number of SSTs involved (Table 1). Both capital and expense dollar savings can be realized by implementation of the TRUEX process. Substantial reductions in expenditures for Hanford Waste Vitrification Plant (HWVP) operation and for fees for geologic disposal of vitrified SST waste contribute

to savings in expense dollars. Savings in capital funds result from eliminating the need to construct additional HWVPs.

Without additional processing steps, application of the TRUEX process to acid dissolved, water-washed sludge could route  $^{90}\text{Sr}$  and uranium to the grout product. In all cases,  $^{137}\text{Cs}$  will, unless removed, be incorporated in grout for subsurface disposal. Comprehensive performance assessments must be performed to fully evaluate environmental impacts of subsurface disposal in engineered vaults, of all or part of the SST inventory of  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , uranium, and other radionuclides.

If necessary, as demonstrated by the results of performance assessments, well-known ion exchange technology can be used to remove  $^{137}\text{Cs}$  from alkaline SST solutions; the concentrated  $^{137}\text{Cs}$  fraction would constitute part of the feed to the HWVP. Precipitation, ion exchange, and solvent extraction processes all appear potentially applicable to removal of  $^{90}\text{Sr}$  from acidic TRUEX process raffinate. The concentrated  $^{90}\text{Sr}$  fraction could also be vitrified in the HWVP.

Hill, O. F., 1970 (August), *Salt Conversion Into Metal (SCIM)*, ARH-1810, Atlantic Richfield Hanford Company, Richland, Washington.

Hodgson, K. M., 1979 (December), *Status of Solids/Liquids Separation Development for Separation and Concentration of Hanford High-Level Defense Waste*, RHO-CD-846, Rockwell Hanford Operations, Richland, Washington.

The separation and concentration process uses water washing and ion exchange to accomplish a chemical separation of the radionuclides from industrial chemicals. The resulting small quantity of highly radioactive wastes is then ready to be incorporated into an immobile waste form. The majority of the radionuclides are contained in the insoluble chemical fraction (sludge) which precipitated from the original liquid waste stored in the underground tanks. The purpose of the sludge washing process is to reduce the volume of the radioactive material that must be immobilized by removing water soluble industrial chemicals from the insoluble sludge and to reduce the concentration of soluble chemicals that tend to degrade the immobile waste forms. This volume reduction results in a substantial cost savings in the storage and disposal process. This savings is realized for all waste forms. Sodium ions also degrade such waste forms as glass, ceramics, bitumen, and concrete. Sulfate, which is removed by washing, degrades glass and ceramic waste forms. The reduction of nitrate by washing is desirable for high-temperature waste forms. The purpose of this document is to report the status of the sludge washing and solids/liquids separation process development activity.

The sludge washing and solids/liquids separation development work has shown that the sodium and sulfate ion concentrations can be reduced to acceptable levels and that solids/liquids separation can be obtained with some of the methods tested. However, further development is necessary.

Gravity thickening is a potential solids/liquids separations method. Separations was obtained when the original synthetic sludge contained 550 ppm of an anionic polyelectrolyte. Additional gravity settling test are required with actual sludges to determine if they will perform the same as the synthetic sludges.

In order to select a solids/liquids separation method, additional studies need to be performed on the many types of separation methods available. This can be accomplished through onsite testing and offsite vendor tests.

Jantzen, C. M., 1988 (November), *Glass Compositions and Frit Formulations Developed for DWPF*, DPST-88-952, Savannah River Laboratory, Savannah River, Georgia.

The document describes a family of glass compositions and frit formulations that have been developed for DWPF. The history of the development work is summarized. The result is a recommendation of a frit for initial DWPF operations which when vitrified with the DWPF waste, over its composition range, will meet the processing and repository acceptance requirements.

Kaser, J. D., 1985 (September), *Removal of Transuranic (TRU) Elements from PUREX Current Acid Waste (CAW) and Neutralized Current Acid Waste*, SD-WM-TA-011, Revision 0, Rockwell Hanford Operations, Richland, Washington.

This study assesses transuranic (TRU) element separation from PUREX neutralized current acid waste (NCAW) generated through 1996. Transuranic element separation reduces the cost of vitrification by reducing the volume of glass produced. In addition to solid-liquid separation, the only promising process for TRU element removal is the TRUEX solvent extraction process. Three facilities were considered for treating the waste by the TRUEX process: 1) B Plant, 2) head end of the PUREX Plant, and 3) The HWVP with added cell space.

The lowest cost alternative is to remove TRU from the waste in the head end of PUREX after implementation of PUREX Facility Modification (PFM) shear/leach decladding. The life cycle cost for this option is estimated at \$173 million, which is \$100 million less than the currently planned alternative of vitrifying all NCAW sludge. Implementation of TRUEX in PUREX would require the loss of a dedicated spare aging waste tank from 1990 to 1993. However, one million gallons of spare aging waste storage space will still be available.

The volume of grout is increased from 3.9 million gallons for the current alternative to 6.9 million gallons for TRUEX process operation, and most of the radiostrontium will end up in the grout rather than in the glass. The maximum concentration of  $^{90}\text{Sr}$ , which can be tolerated in grout disposed of near surface, needs to be determined. If  $^{90}\text{Sr}$  separation is required, the best method of separation must be identified and the cost of  $^{90}\text{Sr}$  separation must be estimated.

Kaser, J. D., and B. A. Higley, 1984 (August), *Alternatives for Treatment of Neutralized Plutonium Finishing Plant Liquid Wastes for Disposal*, SD-WM-ES-038, Revision 0, Rockwell Hanford Operations, Richland, Washington.

The following six alternatives for treatment and disposal of liquid TRU waste from the Plutonium Finishing Plant (PFP) are evaluated:

- Continue current PFP operating mode. Vitrify all sludge for geologic disposal.
- Continue current PFP operating mode. Convert all sludge to grout for geologic disposal.
- Continue current PFP operating mode. Treat all sludge in B Plant for TRU separation and Plutonium recovery.
- Continue current PFP operating mode to 1991. Vitrify pre 1991 sludge. Beginning in 1991 treat waste in PFP for TRU separation and plutonium recovery.
- Continue current PFP operating mode to 1991. Treat all waste for TRU separation and plutonium recovery. Pre 1991 waste is treated in B Plant and post 1991 waste at PFP.
- Initiate TRU separation and Pu recovery at PFP as soon as possible. Treat prior generated waste in B Plant for TRU separation and Plutonium recovery.

The last alternative is the least expensive and recovers up to 410 kg of plutonium, while the first alternative is the most expensive and recovers no plutonium.

Kaser, J. D., B. A. Higley, and M. J. Kupfer, 1983 (June), *Alternatives for Disposal of Hanford Liquids & Sludges Which May Not be Suitable for In Situ Disposal*, SD-WM-ES-012, Revision 0, Rockwell Hanford Operations, Richland, Washington.

The cost of selected options for solidifying and disposing of radioactive waste liquids and sludges are compared. Glass and concrete were the two waste forms considered for geologic disposal. The cost impacts on waste disposal of cesium, strontium and TRU element removal were estimated. By-product and TRU separation costs are not included.

The major findings of this study are:

- Packaging & repository disposal are the most expensive process elements.
- Removal of radiocesium greatly reduces the cost of disposing of the non-TRU portion of the waste.
- Removal of TRU can result in large savings in packaging and disposal costs.

- Repository disposal of glass may be less expensive than repository disposal of concrete."

KEH, 1977 (September), *Final Report, Hanford Defense High-Level Waste Management Studies*, 77-09-RE, Prepared for the U.S. Energy Research and Development Administration by Kaiser Engineers Hanford Company, Richland, Washington.

This document was prepared to provide information on possible alternatives that will be considered for the long-term management of high-level radioactive nuclear waste accumulated as part of the national defense effort at the Hanford Reservation near Richland, Washington. It describes a number of alternatives for retrieval, treatment, and long-term storage of the raw wastes now stored in underground tanks and the treated waste stored in water basins at Hanford. The descriptions include implementation technology, a safety assessment, and preliminary cost estimates. The cost estimates, although useful to compare alternatives, are not of budget quality.

Kaiser Engineers prepared this document in conjunction with a report published by the U. S. Energy Research and Development Administration titled "Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Hanford Reservation, Richland Washington" ERDA document ERDA-77-44. The Kaiser Engineers report is intended to support this ERDA document by providing more detailed information concerning the waste management alternatives discussed in ERDA-77-44.

This document does not take into account either social and public policy issues or the environmental impacts of the alternatives discussed. Instead, the document presents information that is possible to quantify concerning the technology, safety, and costs of waste management alternatives to provide a preliminary basis for discussion and judgement in future decision-making. No selection or recommendation of an alternative for implementation is made in this document. Information contained in this document will be considered in the preparation of a programmatic environmental impact statement and in the selection of waste forms) and storage mode(s) for long-term management of these wastes."

Klem, M. J., J. F. Fletcher, C. E. Golberg, R. D. Gibby, K. A. Giese, F. A. Ruck, J. C. Sonnichsen, D. D. Wanner, N. R. Wing, and K. A. Woodworth, 1990 (June), *Technology Program Plan for Closure of the Single-Shell Tank Operable Units*, WHC-EP-0288, Westinghouse Hanford Company, Richland, Washington.

This Technology Program Plan for Closure of the Single-Shell Tank Operable Units (TPP) provides documentation of the required technology, resources, equipment, program funding, and plans for closure of the six single-shell tank (SST) operable units (OU). The SST OUs comprise treatment, storage, and disposal units (wastes, tanks, and soil contaminated by leaks) and past practice units (ancillary units and soil contaminated by spills). These units are regulated under the Resource Conservation and Recovery Act Section 3005 (e) (interim status permit authority) and Section 3004 (u) (past practices).

A system engineering approach is being used as a management tool to assist in reaching a final disposal decision for the SST OUs. The systems approach is a structured process to define and solve a problem. It is useful for large programs that involve multiple scientific and engineering disciplines and span long time periods. The systems approach ensures that development activities are conducted in an integrated, efficient, thorough, logical, defensible, auditable, and verifiable manner. It will allow the U.S. Department of Energy to meet Hanford Federal Facility Agreement and Consent Order milestones and develop the technology required for the supplemental environmental impact statement for SST waste. The systems engineering for closure of the six single-shell tank OUs is broken down into nine function areas.

These nine functions represent a set of actions that will be permanent throughout the development and implementing phases. This breakdown becomes the framework for planning as the program transits from development to implementation. The functions are divided into main elements or subfunctions and related tasks to provide more detail. Descriptions, special assumptions and constraints, projected costs, and schedules were developed to quantify the requirements and provide a baseline for future planning.

It does not appear economically attractive to vitrify the large amounts (up to 1,400 metric tons in 149 tanks) of uranium isolated by TRUEX process operation with dissolved water-washed SST sludge. If not acceptable for subsurface disposal in grout form, the TRUEX process uranium product could be purified further (e.g., by a tri-*n*-butyl phosphate extraction process), calcined to  $UO_3$ , and stored until it can be satisfactorily used in some part of the U.S. Department of Energy (DOE) nuclear fuel cycle.

Kupfer, M. J., 1987 (March), *Costs and Impacts of Retrieval and Processing of Wastes from Selected Single-Shell Tanks*, RHO-WM-EV-17P, Rockwell Hanford Operations, Richland, Washington.

This report determines the costs of retrieval and processing of wastes from selected Hanford Site underground single-shell tanks and the impacts of retrieval on waste pretreatment operations, grout, and glass production. The assumptions and methods used for determining the costs are consistent with those used for costing the disposal alternatives described in the Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the Hanford Defense Waste Environmental Impact Statement (Rockwell 1985).

Retrieval options were chosen based on projected transuranic element inventory and transuranic element concentration in wastes in single-shell tanks. Retrieval options range from as few as one single-shell tank to as many as 116 tanks. The case of retrieving all 149 single-shell tanks is covered in Rockwell 1985.

Waste is assumed to be removed from single-shell tanks using mechanical retrieval equipment. The feed pretreatment steps include dissolution of the salt cake and separation of the sludge and dissolved salt cake by centrifugation. The sludge is washed with water to assure dissolution of soluble salts. Organic complexants are destroyed in

waste from certain single-shell tanks by ozonization of the dissolved salt cake. The dissolved salt cake is converted to a cementitious grout form for near-surface disposal in concrete vaults. The washed sludge is converted to borosilicate glass for disposal in a deep geologic repository.

Disposal of single-shell tank and double-shell tank waste (both existing and future) is estimated to cost 1.7 billion fiscal year (FY) 1983 dollars for the Reference Alternative disposal option (Rockwell 1985). The Transportable Grout Facility will be able to process single-shell tank waste for selective-retrieval options involving retrieval of waste from up to 61 tanks, as well as the existing and future double-shell tank waste. An additional facility is required if 61 or more tanks are retrieved. New feed tanks and pipelines would be required to accommodate feed from the single-shell tanks. The glass melter will be capable of vitrifying waste from only about five single-shell tanks, in addition to vitrification of the existing and future waste from double-shell tanks. However, this is based on the conservative assumed throughput of only 30 kg of glass per hour and a maximum campaign time of 18 yr (used in Rockwell 1985). (In the present conceptual design both the melter throughput and campaign time can be expanded.)

Kupfer, M. J., 1989 (July 31), *Evaluation of Costs for Selected Retrieval and Processing of Wastes from Single-Shell Tanks*, SD-WM-TI-226, Revision 0, Westinghouse Hanford Company, Richland, Washington.

The costs of retrieval and processing of wastes from selected single-shell tanks (SSTs) and the impacts of retrieval on waste pretreatment operations, grout, and glass production were determined. The assumptions and methods used for determining costs for the selective retrieval options were consistent with those used for costing the disposal alternatives described in the Hanford Defense Waste - Environmental Impact Statement Engineering Data Packages (HDW-EIS EDP). The transuranic (TRU) inventory and TRU concentrations in SSTs were used as the basis for choosing candidate tanks for retrieval.

The following impacts on processing operations were identified:

- B Plant can handle dissolution and sludge washing of waste from 8-14 SSTs based on a maximum operating campaign of 18 years.
- The Transportable Grout Facility can process the maximum volume of SST waste envisioned for the partial retrieval scenarios, as well as the existing and future double-shell tank (DST) waste. However, new feed tanks and pipelines would be required to accommodate feed from the SSTs.
- The glass melter will be capable of vitrifying waste from only about five SSTs in addition to the existing and future waste from DSTs. However, this is based on the conservative HDW-EIS EDP assumed throughput of only 30 kg glass/hr and a maximum campaign time of 18 years.



Kupfer, M. J., A. L. Boldt, and J. L. Buel, 1989 (September), *Process and Facility Options for Pretreatment of Hanford Tank Waste*, SD-WM-TA-015, Revision 0, Westinghouse Hanford Company, Richland, Washington.

The subject report provides an assessment of process and facility options for treating Hanford Site tank waste for immobilization and final disposal. Currently known options for treatment and immobilization of double-shell tank (DST) wastes, new and existing facilities for performing the processing operations, and the timing and capacity of needed feed pretreatment facilities are evaluated. The lower cost processing and facility options that are of reasonable technical certainty are identified. A preferred option is identified that can result in a potential waste disposal program savings of \$500 million. The preferred option involves water washing of neutralized current acid waste (NCAW) sludge in a DST or in the 244-AR Vault (rather than in B Plant), and accelerating implementation of the transuranic extraction (TRUEX) process at B Plant for treatment of follow-on DST wastes. Increasing the vitrification capacity for DST wastes from 45 kg/h to 100 kg/h is also recommended for the preferred option.

Major issues pertaining to both waste processing and facility options, and appropriate development requirements to resolve these issues are identified.

This report provides information that was developed and presented in draft form in fiscal year (FY) 1988. Several follow-on studies have since been performed that addressed key items and recommendations made in this report. The report "Assessment of Double-Shell Tank Waste Pretreatment Options," (Sec 3.3.2) summarizes this information. The conclusions and recommendations in this report have not been updated to incorporate any changes to major assumptions, e.g., those associated with operational schedules, milestones, and costs. Issuance of this report in final form provides detailed background information and bases that support the more recent studies.

Neutralized current acid waste (NCAW) sludge washing will be performed at the 244-AR Vault. This document contains a revised flowsheet description for NCAW pretreatment at AR Vault and B Plant. Upgrades to both facilities are discussed. Time cycles and material balances are calculated.

Kurath, D. E., 1985 (June), *Technology Study for the Pretreatment of Complexant Concentrate*, SD-WM-TA-010, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Several alternatives for the treatment of transuranic (TRU) liquid wastes containing organic complexants were evaluated. TRU removal methods considered were TRUEX solvent extraction and co-precipitation by adding iron nitrate  $[\text{Fe}(\text{NO}_3)_3]$ . The TRUEX process uses an organic extractant to remove TRU and the  $\text{Fe}(\text{NO}_3)_3$  is thought to remove TRU by absorption. The TRU removal/organic destruction alternatives considered were ozonization, oxidation with hydrogen peroxide, high temperature/high pressure, and oxidation in supercritical water. These processes work by destroying the complexing ability of the organics, thereby

allowing the TRU to precipitate. These alternatives were compared against the alternative of direct disposal of the waste in glass. The TRUEX solvent extraction process was found to be the preferred alternative on the condition that the waste streams from this process are compatible with final disposal as grout and glass. The TRUEX process was found to minimize cost, maximize safety and utilize Hanford experience. The TRUEX process also has the flexibility to process other wastes such as existing Plutonium Finishing Plant waste and neutralized current acid waste. Extensive process development is required before this process can be implemented.

For organic destruction the most promising alternatives were found to be oxidation in supercritical water and oxidation with hydrogen peroxide. Extensive process development is required before these processes can be implemented."

Kurath, D. E., 1986 (January), *Technology Program Plan for the Pretreatment of Complexant Concentrate*, SD-WM-TPP-018, Revision 0, Rockwell Hanford Operations, Richland, Washington.

This technical plan describes the work effort to achieve the final disposal of complexant concentrate. Specifically this involves addressing technology for the following:

- Development of TRUEX solvent extraction for TRU removal from the complexant concentrate.
- Determine the need for organic destruction.
- Investigation of organic destruction methods as a contingency action.

Execution of this plan will be performed by Rockwell Hanford Operations, Oak Ridge National Laboratory, Argonne National Laboratory, and Pacific Northwest Laboratory.

Kurath, D. E., and C. J. Yeager, 1987 (May), *Integrated Technology Program Plan for the Treatment of NCRW*, SD-WM-TPP-036, Revision 1, Rockwell Hanford Operations, Richland, Washington.

The NCRW sludge produced as of 1/1/87 has been shown to have a TRU radionuclide level that is too high for direct disposal of the sludge as a grout in near surface vaults. Direct vitrification is not desirable because of a large cost impact. Consequently, a pretreatment method or an alternative disposal method is required for the NCRW sludge. It is also desirable to find a process that can be implemented in PUREX that will produce a low-TRU NCRW sludge that is suitable for disposal in near surface vaults.

The proposed solutions for solving the problem in PUREX center around enhancing the solids removal capability. These methods are:

- addition of flocculating agents
- addition of alternative forms of rare earth
- inertial filtration
- pneumatic hydropulse filtration

The proposed solutions for dealing with the TRU NCRW sludge include:

- blending with other wastes for new surface grout disposal
- grout disposal at WIPP
- pretreatment to remove and concentrate the TRU fraction for vitrification and the low level fraction to near surface grout disposal

This technical program plan provides for the proper integration of chemical processing and waste management tasks to solve the TRU NCRW sludge problem. As part of this effort the following is included: task descriptions; status; cost estimates for unfunded tasks; organizations responsible for tasks; integrated schedule; and key technical decisions.

Lutton, T. W., W. W. Schulz, D. M. Strachan, and L. J. Bollyky, 1980 (March), *Ozonation of Hanford Nuclear Defense Waste*, RHO-SA-98, Rockwell Hanford Operations, Richland, Washington.

High (e.g., 0.1 to 0.5M) concentrations of ethylenediaminetetraacetate (EDTA), 2-hydroxyethylethylenediaminetriacetate (HEDTA), and other organic complexing agents are present in some of the nuclear waste solutions currently stored at the U. S. Department of Energy Hanford Site in southeastern Washington State. Ozonolysis of these alkaline solutions smoothly and efficiently destroys the organic material thereby facilitating ion exchange removal of <sup>99</sup>Sr and other long-lived cationic radionuclides. Successful bench-scale ozonation tests have been performed with both synthetic and actual waste liquids.

Nankani, F. D., 1984 (October), *Hanford Waste Pretreatment Processes*, SD-RE-TI-134, Revision 0, Rockwell Hanford Operations, Richland, Washington.

The purpose of this document is to define the processes required to deliver an acceptable feed to the Hanford Waste Vitrification Plant (HWVP) for vitrification. This document includes the tanks available for storage, the use of existing equipment, and the necessary upgrades to existing equipment in B Plant. B Plant is the designated facility for all pretreatment processes.

These upgrades are necessary if B Plant is to pretreat the various Hanford Site wastes to make them suitable for immobilization to glass. The feed will be prepared from existing and future waste including neutralized current acid waste (NCAW), complexant concentrate (CC), and existing double-shell tank waste that requires vitrification.

The stream compositions and flow diagrams were developed to provide information to feed pretreatment, melter, and grout process design personnel.

Richmond, W. G., 1988 (November 14), *Methods and Data for Use in Determining Source Terms for the Grout Disposal Program*, SD-WM-TI-355, Revision 1, Westinghouse Hanford Company, Richland, Washington.

RHO, 1980 (October), *Technical Status Report on Environmental Aspects of Long-Term Management of High-Level Defense Waste at the Hanford Site*, RHO-LD-139, Rockwell Hanford Operations, Richland, Washington.

This report provides information on the environmental aspects of four alternative methods for long-term management of high-level defense radioactive wastes (HLW) stored at the U.S. Department of Energy (DOE) Hanford Site near Richland, Washington. This information will be used in preparing an environmental impact statement (EIS) on the disposition of Hanford defense waste (HDW). The HDW-EIS is planned for issuance in draft form for public comment and will also address the large amounts of transuranic (TRU) wastes at Hanford. However, this report addresses only the defense HLW.

Since 1944, radioactive wastes have accumulated at DOE's 500-km<sup>2</sup> (570-mi<sup>2</sup>) Hanford Site in southeastern Washington, where nine nuclear reactors have produced nuclear materials for National defense. Today, only one production reactor is still operating, but a large inventory of radioactive high-level waste, the residue from processing the spent fuel to recover plutonium and uranium, remains stored in underground tanks and in metal capsules in water basins. So that this waste will pose no significant threat to the public health and safety, it must be isolated from the biosphere for thousands of years.

This document contains an evaluation of environmental impacts of four alternative methods for long-term management of these HLW. The alternatives range from continuing the present action of storing the waste near the surface of the ground to retrieving the waste and disposing of it deep under ground in a mined geologic repository. The alternatives are:

- A - Near-term geologic disposal of stored waste
- B - Deferred geologic disposal of in-tank waste
- C - In situ disposal of in-tank waste
- D - Continued present action for stored waste

The environmental impacts of the four alternatives are small relative to that radiation received from natural sources or the available natural resources in the earth.

RHO, 1980 (October), *Technical Aspects of Long-Term Management Alternatives for High-Level Defense Waste at the Hanford Site*, RHO-LD-141, Rockwell Hanford Operations, Richland, Washington.

This report provides information on technical aspects of nine alternative methods for long-term management of Hanford Site High-Level Wastes (HLW) (six for in-tank waste and three for encapsulated wastes).

The following conclusions can be drawn for in-tank waste:

- Continued storage in the tanks for an indefinite period of time, either with engineered barriers (in situ disposal, Specific Alternative 7) or without engineered barriers (Specific Alternative 8) is by far less costly than any alternative that requires processing the waste and disposing of it in another location. The next least

expensive alternatives are those incorporating radionuclide concentration which greatly reduces the volume of waste requiring high integrity containerization and storage.

The following conclusion can be drawn for encapsulated waste:

- Continued storage in an onsite near-surface facility (Specific Alternative 11) is more costly than either alternative that requires disposal in a geologic repository due to the cost of surveillance for 250 years.

Routine and accidental releases of radionuclides were calculated and were largest for those alternatives in which in-tank wastes are retrieved and processed for disposal in a geologic repository. On basis of results and insights gained during evaluation of the alternatives described in this report, guidance can be provided to narrow the scope of the technology development program as follows:

- Development efforts for near-term disposal alternatives should focus on processes which reduce the volume of waste to be placed in a repository.
- Research and development efforts should be emphasized in areas relating to in situ (near-surface) disposal of in-tank waste, e.g., the potential for leaving as much waste as possible in tanks to reduce costs and potential radiologic risk while meeting criteria for safe storage and disposal of chemical and radioactive wastes.

RHO, 1983 (December), *Hanford Waste Management Technology Plan*, RHO-WM-PL-9, Rockwell Hanford Operations, Richland, Washington.

RHO, 1985 (December), *Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the Hanford Defense Waste - Environmental Impact Statement*, RHO-RE-ST-30 P, Rockwell Hanford Operations, Richland, Washington.

This document provides the engineering bases for the development of the Hanford Defense Waste-Environmental Impact Statement. In compliance with the National Environmental Policy Act process and, more specifically, the detailed scope prepared for the Hanford Defense Waste-Environmental Impact Statement, four waste disposal alternatives are identified: geologic disposal; in-place stabilization and disposal; continued storage (no disposal action); and the reference alternative. For each disposal alternative, the following six waste type classifications are addressed: existing tank waste, transuranic-contaminated soil sites (cribs and reverse wells), pre-1970 transuranic buried solid waste sites, retrievably stored and newly generated solid transuranic waste, strontium and cesium capsules, and future tank waste. The disposal alternatives are presented as options for the disposal of each waste type. Data regarding structures, site locations, and inventories for each waste class are provided, and are followed by a description of various technologies applied for implementing the disposal alternatives. Data associated with the resulting impacts (resources consumed, manpower used, emissions, and costs) are tabulated

according to the waste class/alternative matrix. This information was used during the preparation of the Hanford Defense Waste-Environmental Impact Statement to develop socioeconomic analyses, accident scenarios, dose estimates, and waste release or migration evaluations.

RHO, 1986 (November), *Integrated Grout Management Plan*, RHO-RE-PL-12P, Rockwell Hanford Operations, Richland, Washington.

RHO, 1987 (February), *Engineering Support Data Update for the Hanford Defense Waste - Environmental Impact Statement*, RHO-RE-ST-30 ADD P, Rockwell Hanford Operations, Richland, Washington.

This document provides updated engineering support data for development of an environmental impact statement for Hanford defense, high-level, transuranic, and tank wastes. This document should be used in conjunction with the original engineering support data entitled Hanford Defense Waste - Environmental Impact Statement. The update data are intended to reflect data and information gathered since 1983, and are current to January 1987. Updated data include inventories, site descriptions, engineering methodologies for retrieval of single-shell tank waste, and facilities descriptions and costs. Errata for the original engineering data is also included as an appendix.

Richardson, G. L., 1980 (November), *Deferred Processing of Hanford High-Level Wastes*, HEDL-TME 80-48, Hanford Engineering Development Laboratory, Westinghouse Hanford Company, Richland, Washington.

This document was prepared to provide detailed engineering and environmental information on possible alternatives that may be used for deferred retrieval and disposal of high-level in-tank wastes derived from the nuclear defense program at Hanford.

An evaluation of the incentives for deferred processing of high-level waste (HLW) shows that the implementation costs (using trust fund annuity financing) and potential radiological health effects to the public decrease with time for the first 200 years of deferral but tend to level off after about 200 to 300 years. Thus, a deferral period of 250 years was selected for detailed evaluation of the deferred retrieval alternatives. At this time, the gamma activity will have decreased to the point that the waste can be handled and disposed of as a low-gamma-level transuranic (TRU) waste.

To parallel Rockwell's near-term retrieval and processing scenarios, other processing concepts involving both radionuclide concentration and bulk disposal were evaluated. A bulk fused salt process was selected as the reference process for Specific Alternative 5 for onsite disposal in a basalt repository, and a modified radionuclide concentration/vitrification (RC/V) process was selected for Specific Alternative 6 for offsite disposal in a bedded salt repository. These processing scenarios are considered to adequately bracket the range of impacts that may be incurred for deferred processing.

Rizzo, A. J., 1989 (External Letter to R. M. Bernero, U.S. Nuclear Regulatory Commission, Washington, D.C., March 6, 1989), U.S. Department of Energy, Richland Operations Office, Richland, Washington.

The classification of the fraction of double-shell tank waste that will be disposed of near-surface as grout is addressed. An overall radionuclide material balance for all Hanford wastes is presented. Based on consideration of alternative separation processes, and to meet the suggested criteria of segregating the largest practical amount of activity, removal of the least 95% of the Cs-137 from complexant concentrate waste is proposed.

Schulz, W. W., 1980 (January), *Cyclohexanone Solvent Extraction of  $^{99}\text{TcO}_4$  From Alkaline Nuclear Waste Solutions*, RHO-SA-123, Rockwell Hanford Operations, Richland, Washington.

Laboratory scale tests were performed to evaluate a solvent extraction process for removing  $\text{TcO}_4^-$  (pertechnetate anion) from Hanford alkaline waste solutions using cyclohexanone as the extractant. Distribution coefficients of  $\text{TcO}_4^-$  between aqueous alkaline nitrate feed and cyclohexanone are high enough to permit satisfactory countercurrent engineering-scale extraction of  $^{99}\text{Tc}$ . Technetium can be removed from cyclohexanone extracts by simply stripping with water, although phase disengaging problems were encountered during water stripping operations on a laboratory scale. Stripping tests in pulse columns and/or centrifugal contactors are needed to determine the magnitude of the phase disengaging problem and to find suitable remedies.

Schulz, W. W., 1980 (January), *Removal of Radionuclides from Hanford Defense Waste Solutions*, RHO-SA-51, Rockwell Hanford Operations, Richland, Washington.

The Hanford high-level defense wastes are characterized by their large volume (~190 000 m<sup>3</sup>) and varying content of inert and radioactive constituents. The water-soluble portion (~140 000 m<sup>3</sup>) of these wastes, which consists mainly of  $\text{NaNO}_3$ ,  $\text{NaAl}(\text{OH})_4$ ,  $\text{Na}_2\text{CO}_3$  and other sodium salts, contains a few milligrams of long-lived ( $t_{1/2} \geq 10$  years) radionuclides per 1000 kilograms. There is probable economic incentive for long-term management of Hanford defense wastes to partition them into a small volume of highly radioactive material requiring high integrity immobilization and storage and a much larger fraction of low-level (e.g., <10 nCi/g) waste which can be economically and safely stored in bulk form. To aid in achieving this latter objective, an integrated series of aqueous separations processes (precipitation, ion exchange, and solvent extraction methods) was designed to remove  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , actinides, other multivalent cationic fission and activation products, and  $^{99}\text{Tc}$  from the water-soluble wastes. Results of generally satisfactory laboratory-scale tests of radionuclide removal technology with actual Hanford wastes are described.

Schulz, W. W., and L. D. McIsaac, 1975 (August), *Removal of Actinides from Nuclear Fuel Reprocessing Waste Solutions with Bidentate Organophosphorus Extractants*, ARH-SA-217, Atlantic Richfield Hanford Company, Richland, Washington.

The neutral bidentate organophosphorus reagents DBDECMP (dibutyl-N,N-diethylcarbamylnmethylenephosphonate) and its dihexyl analogue DHDECMP are candidate extractant for removal of actinides from certain acidic waste streams produced at the U. S. Energy Research and Development

Administration's Hanford and Idaho Falls sites. Various chemical and physical properties including availability, cost, purification, alpha radiolysis, and aqueous phase solubility of DBDECMP and DHDECMP are reviewed. A conceptual flowsheet employing a 15% DBDECMP (or DHDECMP)-CCl<sub>4</sub> extractant for removal (and recovery) of Am and Pu from Hanford's Plutonium Reclamation Facility acid waste stream (CAW solution) was successfully demonstrated in laboratory-scale mixer-settler tests; this extraction scheme can be used to produce an actinide-free waste. A 30% DBDECMP-xylene flowsheet is being tested at the Idaho Falls site for removal of U, Np, Pu, and Am from Idaho Chemical Processing Plant first-cycle high-level raffinate to produce an actinide-free (<10 nCi alpha activity/gram) waste.

Schulz, W. W., M. J. Kupfer, and J. P. Sloughter, 1983 (December), *Evaluation of Process and Facility Options for Treatment of Double-Shell Tank Wastes*, SD-WM-ES-023, Revision 0, Rockwell Hanford Operations, Richland, Washington.

An engineering study was performed to define and evaluate options for preparing existing and future double-shell tank wastes for immobilization (glass or grout); preferred feed preparation processes, facilities, and schedules were determined. Three preferred flowsheets for preparing immobilization facility feeds from six candidate wastes [Current Acid Wastes (CAW), Neutralized Current Acid Waste (NCAW), Double-Shell Slurry (DSS), Complexant Concentrate (CC), Cladding Removal Waste (CRW), and Plutonium Finishing Plant (PFP) Wastes] were derived by applying screening criteria to an initial 80 process options.

Three different facilities [B Plant, Expanded B Plant Immobilization Pilot Plant (BIPP) and New Stand Alone Facility] were evaluated for performing the waste preparation steps. Costs of conducting the preferred sequence of feed preparation operations in each of the three facilities were estimated for facility startup dates in the period 1986 to 2000.

Based upon analysis and evaluation of the significant findings of this study, the following facility selection and deployment schedule for feed preparation and immobilization facilities are recommended:

- Upgrade the existing B Plant for FY 1986 start of feed preparation operations for CRW, CC, and if necessary DSS wastes.
- Complete design and construction of a transportable grout facility to start immobilization and near-surface disposal of candidate wastes (DSS, Customer Wastes) in FY 1986.
- Complete design and construction of the BIPP facility to bring it on line in FY 1991.

Schulz, W. W., M. M. Beary, S. A. Gallagher, B. A. Higley, R. G. Johnston, F. M. Jungfleisch, M. J. Kupfer, R. A. Palmer, R. A. Watrous, and G. A. Wolf, 1980 (September), *Preliminary Evaluation of Alternative Forms for Immobilization of Hanford High-Level Defense Wastes*, RHO-ST-32, Rockwell Hanford Operations, Richland, Washington.



The document presents a preliminary evaluation of solid waste forms for immobilization of Hanford high-level radioactive defense wastes. Nineteen waste forms were evaluated and compared to determine their applicability and suitability for immobilization of Hanford salt cake, sludge, and residual liquid. Waste forms were evaluated and ranked on the basis of weighted ratings of seven waste form and seven process characteristics. Borosilicate glass was ranked among the first three choices for fixation of all Hanford HLW.

Schulz, W. W., M. M. Beary, R. A. Watrous, R. G. Johnston, and J. V. Panesko, 1982 (June), *Inventories and Technology for Recovery of Americium, Promethium, Rhodium, and Palladium Values at Hanford: A Preliminary Assessment*, RHO-LD-170, Rockwell Hanford Operations, Richland, Washington.

Inventories and current economic worths of  $^{241}\text{Am}$ ,  $^{147}\text{Pm}$ , and stable rhodium and palladium in existing Hanford wastes and in future Hanford PUREX Plant high-level waste were calculated and are presented. "Conceptual process flowsheets for recovery of a crude americium and promethium fraction either in the PUREX Plant (via DHDECMP solvent extraction) or in B Plant (via currently used HDEHP solvent extraction) are presented. A pyrometallurgical process under development at Pacific Northwest Laboratory appears technically suitable for recovery of a crude rhodium and palladium fraction from vitrified acidic (and possibly alkaline) future PUREX high-level waste. A conceptual amine solvent extraction process that involves head-end removal of aluminum and  $^{99}\text{Tc}$  is discussed for recovery of rhodium and palladium values from existing highly alkaline waste. Major research and development tasks needed to implement and/or support recovery of  $^{241}\text{Am}$ ,  $^{147}\text{Pm}$ , rhodium, and palladium from future PUREX high-level waste and/or existing wastes are listed.

Schwoebel, R. L., and C. J. Northrup, 1978 (November), *Proceedings of the Sandia Laboratories Workshop on the Use of Titanate Ion Exchangers for Defense Waste Management*, SAND78-2019, Sandia National Laboratory, Albuquerque, New Mexico.

This workshop convened some of the principal technical participants involved in programs for the stabilization of tank-stored defense wastes at Savannah River Laboratories, the Hanford Reservation, and Nuclear Fuel Services at West Valley, New York. The purpose of the workshop was to discuss baseline objectives and decontamination processes currently planned and/or being investigated by each facility, review studies at Rockwell and Sandia Laboratories of the scientific and engineering applicability of a generic family of inorganic ion exchangers to waste decontamination, and identify future research and development activities required to implement use of these ion exchangers in full-scale decontamination.

The applicability of inorganic titanate ion exchangers to a wide variety of waste management applications, was reviewed. Since 1975, the research on these materials has been directed toward solving the problem of the defense waste decontamination. In a joint program with the Atlantic Richfield Hanford Company, Sandia Laboratories investigated the possibility of efficiently extracting the multivalent ions

(principally  $^{90}\text{Sr}$ ) from the basic, high-salt content defense wastes. These studies indicate that Sr and actinides can be removed from salt cake to such an extent that the total residual activity is  $\sim 10 \text{ n Ci/gm}$ .

A feature of these exchangers that stimulated some interest were experiments indicating that the material could be efficiently eluted. Effective elution could significantly impact the flow sheets by decreasing costs, down time, and operational complexity. It was recommended that additional experiments be performed to detail the elution properties. It was also recommended that the ion exchange properties be investigated to determine the temperature interval over which this material can be processed and still retain its high affinity.

WHC, 1990 (January), *Assessment of Double-Shell Tank Waste Pretreatment Options*, WHC-SP-0464, Revision 1, Westinghouse Hanford Company, Richland, Washington.

Some Hanford Site liquid and solid wastes stored in double-shell tanks (DST) must be pretreated before final disposal in cementitious grout or glass forms. The current baseline waste management plan calls for necessary pretreatment operations to be performed in the upgraded B Plant facility. In addition to the viability of B Plant for pretreatment of DST waste, a comprehensive consideration and examination of alternative facilities, including B Plant, for performing required pretreatment operations was made. A key step in evaluation of the options involved determination of the viability of the existing B Plant facility for the waste pretreatment mission, and the 244-AR Vault for waste lag storage and sludge washing operations.

No issues were found that would prevent B Plant or the 244-AR Vault from completing the pretreatment missions. The need for some additional facility upgrades was identified. With these upgrades the facilities can be brought to a condition that will comply with DOE design criteria, safety, and environmental orders.

Three alternative process and facility strategies were developed and compared. One option (Option B) with sludge washing in the 244-AR Vault and early TRUEX process operations in B Plant resulted in

significant cost savings compared to the other options examined. Key studies were proposed to support final approval of the preferred strategy.

Winters, W. I., 1981 (June), *Effect of pH on the Destruction of Complexants with Ozone in Hanford Nuclear Waste*, RHO-SA-203, Rockwell Hanford Operations, Richland, Washington.

Chemical processing of nuclear waste at Hanford has generated some waste solutions with high concentration (0.1 to 0.5M) of N-(hydroxyethyl)-ethylenediaminetriacetic acid (HEDTA), ethylenediaminetetraacetic acid (EDTA), and other organic complexing agents. These complexants must be destroyed because they affect radionuclide migration in soils, waste concentration, radionuclide removal, and other waste storage and processing considerations. Previous studies on actual waste solutions demonstrated that pre-

ozonation of the alkaline waste significantly improved radionuclide removal. A series of bench-scale experiments using synthetic waste has been performed to determine the optimum pH for most efficient ozone destruction of EDTA. Ozonation of EDTA in synthetic waste was carried out over the pH range of 1 to 14. Potential catalytic materials were examined at different pH levels. The EDTA-ozone reaction rates and stoichiometric requirements were compared and evaluated for the varying conditions.

Wong, J. J., 1989 (October), *244-AR Conceptual Flowsheet for Processing of NCAW*, WHC-SE-WM-TI-396, Revision 0, Westinghouse Hanford Operations, Richland, Washington.

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APPENDIX A

NEW WASTE GENERATORS' FACILITY DESCRIPTIONS,  
TYPES OF WASTE, AND WASTE MINIMIZATION

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## CONTENTS

1.0	NEW WASTE GENERATORS IN 100-N AREA . . . . .	1-1
1.1	DESCRIPTION OF FACILITY AND TYPES OF WASTES GENERATED . . . . .	1-1
1.1.1	Description of Facility . . . . .	1-1
1.1.2	Types of Waste Generated . . . . .	1-1
1.2	WASTE TREATMENT ACTIVITIES . . . . .	1-2
1.2.1	Past Waste Treatment Activities . . . . .	1-2
1.2.2	Present Waste Treatment Activities . . . . .	1-2
1.3	WASTE MINIMIZATION SUMMARY . . . . .	1-3
2.0	NEW WASTE GENERATORS IN THE 300 AREA . . . . .	2-1
2.1	DESCRIPTION OF FACILITIES AND TYPES OF WASTES GENERATED . . . . .	2-1
2.1.1	324 Chemical Engineering Laboratory . . . . .	2-1
2.1.2	325 Radiochemistry Laboratory . . . . .	2-1
2.1.3	326 Materials Technology Laboratory . . . . .	2-2
2.1.4	327 Postirradiation Laboratory . . . . .	2-3
2.1.5	329 Physics Science Laboratory . . . . .	2-3
2.1.6	3720 Building . . . . .	2-4
2.1.7	331 Life Sciences Laboratory . . . . .	2-4
2.1.8	340 Facility . . . . .	2-5
3.0	NEW WASTE GENERATORS AT THE 400 AREA . . . . .	3-1
3.1	DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED . .	3-1
3.2	GENERATION OF TANK WASTES IN THE 400 AREA . . . . .	3-1
3.3	TANK WASTE MINIMIZATION AT THE FAST FLUX TEST FACILITY AND MAINTENANCE AND STORAGE FACILITY . . . . .	3-1
3.4	FUTURE TANK WASTE GENERATED AS A RESULT OF THE FAST FLUX TEST FACILITY SHUTDOWN OPTION . . . . .	3-2
4.0	TANK FARMS . . . . .	4-1
4.1	DESCRIPTION OF THE FACILITIES AND TYPES OF WASTE GENERATED . . . . .	4-1
4.1.1	Single-Shell Tanks . . . . .	4-1
4.1.2	Double-Shell Tanks . . . . .	4-1
4.1.3	Solutions Added to Double-Shell Tanks by Tank Farm Operation . . . . .	4-2
4.2	WASTE MINIMIZATION ACTIVITIES . . . . .	4-4
4.2.1	Past Waste Minimization Activities . . . . .	4-4
4.2.2	Present Waste Minimization Activities . . . . .	4-4
4.3	SUMMARY . . . . .	4-5
5.0	EVAPORATORS . . . . .	5-1
5.1	DESCRIPTION OF EVAPORATOR FACILITIES . . . . .	5-1
5.2	TYPES OF WASTE GENERATED . . . . .	5-1
5.3	WASTE TREATMENT ACTIVITIES . . . . .	5-2
5.3.1	Past Waste Treatment Activities . . . . .	5-2
5.3.2	Present Waste Treatment Activities . . . . .	5-2
6.0	PLUTONIUM FINISHING PLANT . . . . .	6-1
6.1	PLUTONIUM FINISHING PLANT FACILITY DESCRIPTION . . . . .	6-1
6.2	WASTE TYPE DESCRIPTION . . . . .	6-1

## CONTENTS (cont.)

6.3	WASTE MINIMIZATION ACTIVITIES . . . . .	6-1
6.3.1	Implementation of TRUEX at the Plutonium Finishing Plant . . . . .	6-1
6.3.2	Hanford Private Sector Participation Conference . . . . .	6-2
6.3.3	Plutonium Reclamation Facility Process Modification . . . . .	6-2
6.3.4	Segregation of Waste at the RMC Line . . . . .	6-3
6.4	PLUTONIUM RECLAMATION FACILITY WASTE MINIMIZATION PLAN . . . . .	6-3
6.4.1	Purpose . . . . .	6-3
6.4.2	Employee Training . . . . .	6-3
6.4.3	New Projects and Designs . . . . .	6-3
7.0	PUREX PLANT . . . . .	7-1
7.1	DESCRIPTION OF THE FACILITY . . . . .	7-1
7.2	TYPES OF WASTE GENERATED . . . . .	7-1
7.3	WASTE MINIMIZATION . . . . .	7-1
7.3.1	Past Waste Minimization and Pretreatment Activities . . . . .	7-1
7.3.2	Present Waste Minimization and Pretreatment Activities . . . . .	7-2
7.4	WASTE MINIMIZATION AND PRETREATMENT PLAN SUMMARY . . . . .	7-2
8.0	B PLANT . . . . .	8-1
8.1	DESCRIPTION OF B PLANT AND TYPE OF WASTES GENERATED . . . . .	8-1
8.1.1	Description of Facility . . . . .	8-1
8.1.2	Types of Waste Generated at B Plant . . . . .	8-1
8.2	WASTE TREATMENT ACTIVITIES . . . . .	8-1
8.2.1	Past Waste Treatment Activities and Process Flowsheets . . . . .	8-1
8.2.2	Present Waste Treatment Activities and Process Flowsheets . . . . .	8-2
9.0	S PLANT . . . . .	9-1
9.1	DESCRIPTION OF FACILITIES AND TYPES OF WASTE . . . . .	9-1
9.1.1	Description of Facility . . . . .	9-1
9.1.2	Description of Waste . . . . .	9-1
9.2	WASTE MINIMIZATION . . . . .	9-4
10.0	T PLANT . . . . .	10-1
10.1	DESCRIPTION OF FACILITY AND TYPE OF WASTE GENERATED . . . . .	10-1
10.1.1	Description of Facility . . . . .	10-1
10.1.2	Types of Waste Generated . . . . .	10-1
10.2	WASTE MINIMIZATION ACTIVITIES . . . . .	10-1
11.0	HANFORD WASTE VITRIFICATION PLANT . . . . .	11-1
11.1	DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED . . . . .	11-1
11.1.1	Description of Facility . . . . .	11-1
11.1.2	Types of Waste Generated . . . . .	11-1



CONTENTS (cont.)

11.2	WASTE MINIMIZATION ACTIVITIES . . . . .	11-1
11.2.1	Process Waste . . . . .	11-2
11.2.2	Decontamination Waste . . . . .	11-2
11.2.3	Maintenance Waste . . . . .	11-3
11.2.4	Miscellaneous Waste . . . . .	11-3
11.2.5	Nonprocess Waste . . . . .	11-3
12.0	GROUT TREATMENT FACILITY . . . . .	12-1
12.1	DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED . .	12-1
12.1.1	Description of Facility . . . . .	12-1
12.1.2	Type of Waste Generated . . . . .	12-1
12.2	WASTE MINIMIZATION ACTIVITIES . . . . .	12-1
12.2.1	Employee Training . . . . .	12-1
12.2.2	Employee Participation and Incentive Program . .	12-2
12.2.3	New Projects and Designs . . . . .	12-2

LIST OF FIGURES

7-1	Simplified PUREX Plant Flowsheet . . . . .	7-3
7-2	Rare Earth Precipitation Process . . . . .	7-4
7-3	PUREX Plant Ammonia Waste Handling--Before Ammonia Destruction .	7-5
7-4	PUREX Plant Ammonia Waste Handling--With Ammonia Destruction . .	7-6
9-1	Concentration of REDOX Complex Waste . . . . .	9-3

LIST OF TABLES

2-1	The Average Composition of the Waste Handled in the 340 Facility during Fiscal Year 1989 . . . . .	2-6
4-1	Chronology of the Double Shell-Tank Construction . . . . .	4-2
9-1	REDOX Complex Waste Composition . . . . .	9-2
10-1	Waste Transfers to Tank Farms . . . . .	10-2

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## LIST OF TERMS

ALC	air-lift circulator
CAW	current acid waste
CP	concentrated phosphate
CRW	cladding removal waste
DMF	Dry Materials Facility
DOE	U.S. Department of Energy
DOE-RL	U.S. Department of Energy-Richland Operations Office
DSS	double-shell slurry
DSSF	double-shell slurry feed
DST	double-shell tank
EPA	U.S. Environmental Protection Agency
FFTF	Fast Flux Test Facility
FY	fiscal year
GDF	Grouted Waste Disposal Facility
GPF	Grout Processing Facility
GTF	Grout Treatment Facility
HLW	high-level waste
HWVP	Hanford Waste Vitrification Plant
IEM	Interim Examination and Maintenance
LLW	low-level waste
MAP	mixed activation products
MASF	Maintenance and Storage Facility
MFP	mixed fission products
NCAW	neutralized current acid waste
NCRW	neutralized cladding removal waste
NPH	normal paraffin hydrocarbon
OWW	organic wash waste
PFP	Plutonium Finishing Plant
ppm	parts per million
PRF	Plutonium Reclamation Facility
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
RLWS	Radioactive Liquid Waste System
RMC	Remote Mechanical C
RPS	retention process sewer
SST	single-shell tank
SWL	saltwell liquor
TBP	tributylphosphate
TOC	total organic carbon
Tri-Party Agreement	<i>Hanford Federal Facility Agreement and Consent Order</i>
TRU	transuranic
TRUEX	transuranic extraction
WESF	Waste Encapsulation Storage Facility
Westinghouse Hanford	Westinghouse Hanford Company
WIPP	Waste Isolation Pilot Plant

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## 1.0 NEW WASTE GENERATORS IN 100-N AREA

### 1.1 DESCRIPTION OF FACILITY AND TYPES OF WASTES GENERATED

#### 1.1.1 Description of Facility

The principal facility in the 100-N Area is the dual-purpose N Reactor, which was designed to produce special nuclear materials and steam for the generation of electricity. Support facilities for N Reactor include a water-filled fuel storage basin and decontamination systems for the reactor and the fuel storage basin.

#### 1.1.2 Types of Waste Generated

Historically, N Reactor activities have generated between 300,000 and 700,000 gal of tank waste annually. This waste has been primarily of three types:

1. N Reactor decontamination waste
2. Ion-exchange regeneration waste
3. Sand filter backwash.

**1.1.2.1 N Reactor Decontamination Waste.** The N Reactor decontamination waste was produced during periodic reactor decontamination with Turco 4512-A\* detergent used for decontamination. The resulting waste contained significant quantities of trisodium phosphate and lesser quantities of various complexants and inhibitors.

The waste was neutralized to a pH of 7+ before it was transferred to a receiving tank in the 200 East Area by rail tank cars. The N Reactor decontamination waste (phosphate waste) was stored and treated in Evaporator 242-A. The interim storage product for phosphate waste is termed concentrated phosphate waste.

**1.1.2.2 Ion-Exchange Regeneration Waste.** The ion-exchange regeneration waste was produced during the regeneration of the ion-exchange resins used to remove radionuclides from the 105-N spent fuel storage basin water. Sulfuric acid was used to regenerate the cation exchange resin, and sodium hydroxide was used to regenerate the anion exchange resin. Extensive rinsing produced a dilute sodium sulfate waste solution.

The sulfate waste was adjusted to a pH of 7+ by the addition of sodium hydroxide before shipment to the 200 East Area. The sulfate waste was blended with other dilute, noncomplexed wastes for treatment in Evaporator 242-A to produce double-shell slurry (DSS) for interim storage.

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\*Turco 4512-A is a trademark of the Purex Corporation.

**1.1.2.3 Sand Filter Backwash.** A sand filter is used to remove entrained solids from the fuel storage basin water before treatment by ion exchange. The sand filter backwash is primarily an inorganic sludge generated during periodic filter flushing to remove accumulated solids.

The sludge was transferred as a dilute slurry into tank cars where it was mixed with the ion-exchange regeneration waste for shipment to the 200 East Area.

## 1.2 WASTE TREATMENT ACTIVITIES

### 1.2.1 Past Waste Treatment Activities

Past waste treatment activities were centered on Evaporator 242-A where the volume of the 100-N wastes was reduced for storage in the 200 Areas tanks. More recently the concentrated phosphate-sulfate wastes were treated by conversion into a cementitious grout for disposal in a near-surface concrete vault.

#### Phosphate waste

NaOH	0.010 <u>M</u>
NaNO <sub>2</sub>	0.014 <u>M</u> ----->EVAP----->
Na <sub>3</sub> PO <sub>4</sub>	0.363 <u>M</u>
Volume	1.00 gal

#### Concentrated phosphate (CP)

NaOH	0.02 <u>M</u>
NaNO <sub>2</sub>	0.03 <u>M</u>
Na <sub>3</sub> PO <sub>4</sub>	0.72 <u>M</u>
Volume	0.5 gal

#### Sulfate waste

NaOH	0.01 <u>M</u>
NaNO <sub>2</sub>	0.01 <u>M</u> ----->EVAP----->
Na <sub>2</sub> SO <sub>4</sub>	0.04 <u>M</u>
Volume	1.0 gal

#### Double-shell slurry feed (DSSF)

NaOH	0.50 <u>M</u>
NaNO <sub>2</sub>	0.50 <u>M</u>
Na <sub>2</sub> SO <sub>4</sub>	2.00 <u>M</u>
Volume	0.02 gal

#### DSSF

NaOH	0.05 <u>M</u>
NaNO <sub>2</sub>	0.05 <u>M</u> ----->EVAP----->
Na <sub>2</sub> SO <sub>4</sub>	2.00 <u>M</u>
Volume	0.02 gal

#### DSS

NaOH	1.00 <u>M</u>
NaNO <sub>2</sub>	1.00 <u>M</u>
Na <sub>2</sub> SO <sub>4</sub>	4.00 <u>M</u>
Volume	0.01 gal

### 1.2.2 Present Waste Treatment Activities

Present waste treatment activities have ceased as a result of the shutdown of Evaporator 242-A. Fuel is still stored in the fuel storage basin and one additional ion-exchange regeneration is planned, which will produce approximately 36,000 gal of waste between April 1990 and June 1991. No additional tank waste generation is planned.

### 1.3 WASTE MINIMIZATION SUMMARY

Previous waste minimization of 100-N waste consisted of volume reduction through the use of Evaporator 242-A. With the shutdown of N Reactor, the generation of tank waste has essentially ceased, except for a remaining 36,000 gal which have yet to be generated through facility layup activities.

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## 2.0 NEW WASTE GENERATORS IN THE 300 AREA

### 2.1 DESCRIPTION OF FACILITIES AND TYPES OF WASTES GENERATED

In the 300 Area, tank waste is generated in seven different laboratory facilities and stored in the 340 Storage Facility until a sufficient waste volume has been accumulated to warrant its shipment to the tank farms for storage, any necessary treatment, and ultimate disposal. Descriptions of the seven individual laboratory facilities, the 340 Facility, and their individual waste streams are presented in the following sections. A composite analysis of the tank waste generated in the 300 Area is included in the discussion of the 340 Facility.

#### 2.1.1 324 Chemical Engineering Laboratory

Most of the 324 Chemical Engineering Laboratory is occupied by two groups of shielded hot cells and their service and operating galleries. Liquid wastes that are produced during the operation of these hot cell facilities drain into a collection tank located in the 324 Laboratory. The accumulated waste is pumped through the Radioactive Liquid Waste System (RLWS) line to the 340 Facility for temporary storage before transfer by tank car to the tank farms.

The waste generated by the operation of the 324 Laboratory hot cells is generally water that has been contaminated with radioactive materials as a result of being used to clean and rinse contaminated equipment. Other wastes generated in the facility include condensate from the drying solid waste that is being prepared for disposal. A description of the amount and type of waste that is produced in the 324 Laboratory in a typical year follows:

- Volume--1 kgal/yr
- Chemical Composition--water
- Predominant Radionuclides--cesium-137 ( $^{137}\text{Cs}$ ) and strontium-90 ( $^{90}\text{Sr}$ ) with mixed fission products (MFP) and mixed activation products (MAP).

#### 2.1.2 325 Radiochemistry Laboratory

The 325 Radiochemistry Laboratory is a multipurpose laboratory facility with two different sets of hot cells and several analytical laboratories.

The process research hot cells located in the east wing of the 325A Building are used to handle highly radioactive materials for a variety of processes and tests. The inorganic waste produced in the cells generally consist of rinse water and dissolved irradiated fuel sample sections. The waste generated in the 325A Building drains to a collection tank in that facility and, after arrangements have been made, is steam jetted to the RLWS line for accumulation in the 340 Facility. Process research hot cells are

used to extrude and blend core samples from the tank farms. A description of the waste that will be generated in the process research hot cells is as follows:

- Volume--1 kgal/yr
- Chemical Composition--inorganic acid
- Predominant Radionuclides--cerium-144 ( $^{144}\text{Ce}$ ), cobalt-60 ( $^{60}\text{Co}$ ), cesium-134 ( $^{134}\text{Cs}$ ),  $^{137}\text{Cs}$ , and ruthenium-106 ( $^{106}\text{Ru}$ ) with MFP and MAP.

The hot cells in the 325B Building are used to dissolve fuel components and other solids in acid before chemical analysis. The waste that is generated in these hot cells is primarily rinse water, and is only slightly radioactive. These hot cells drain to a small collection tank that is float-switch activated and pumps to the RLWS line and the 340 Facility. A description of the waste generated in the 325B Building cells follows:

- Volume--0.5 kgal/yr
- Chemical Composition--water
- Predominant Radionuclides-- $^{144}\text{Ce}$ ,  $^{60}\text{Co}$ ,  $^{134}\text{Cs}$ ,  $^{137}\text{Cs}$ , and  $^{106}\text{Ru}$  with MFP and MAP.

The analytical laboratory waste generated in the 325 Building is put directly into RLWS drains. Most of the waste is generated from fuel rod analysis. A general description of the waste produced from laboratory analytical work follows:

- Volume--1 kgal/yr
- Chemical Composition--inorganic analytical waste
- Predominant Radionuclides-- $^{144}\text{Ce}$ ,  $^{60}\text{Co}$ ,  $^{134}\text{Cs}$ ,  $^{137}\text{Cs}$ , and  $^{106}\text{Ru}$  with MFP and MAP.

### 2.1.3 326 Materials Technology Laboratory

Most of the work performed in the 326 Materials Technology Laboratory involves the study of metallurgical, chemical, and physical behavior of reactor components and fuel materials. Two laboratories in the facility generate tank waste that is sent to the 340 Facility via RLWS piping.

The metallography laboratory prepares metal samples to be photographed. Several types of polishing media and etching solutions are used to expose the metal crystalline structure for photographing. A general description of the waste from the metallography laboratory follows:

- Volume--1 kgal/yr

- Chemical Composition--water mixed with trace amounts of aluminum oxide, colloidal silicon, diamond paste, and silicon carbide polishing compounds
- Predominant Radionuclides--uranium-235 ( $^{235}\text{U}$ ).

The second laboratory, where radioactive waste is generated, is used to prepare metal coupons for survey in an electron microscope. The coupons are prepared by washing them in several different acids baths. A general description of the waste that is generated in this section of the 326 Building follows:

- Volume--0.2 kgal/yr
- Chemical Composition--dilute perchloric and acetic acids and isobutyl alcohol
- Predominant Radionuclides--radioactive metals.

#### 2.1.4 327 Postirradiation Laboratory

The 327 Postirradiation Testing Laboratory is used for destructive and nondestructive examination of irradiated reactor fuel and structural materials. These examinations and the associated testing are carried out in 12 shielded cells, several of which drain to RLWS piping. The cell drains are filtered to prevent solids from entering the RLWS piping and 340 Facility tanks. Most of the waste is generated during grinding and cutting operations, performed on irradiated fuels and materials, and when the equipment in the cells are cleaned and rinsed. A general description of the waste that is generated by the 327 Laboratory follows:

- Volume--10 kgal/yr
- Chemical Composition--water mixed with decontamination materials
- Predominant Radionuclides-- $^{144}\text{Ce}$ ,  $^{137}\text{Cs}$ , and  $^{90}\text{Sr}$ .

#### 2.1.5 329 Physics Science Laboratory

The 329 Physics Science Laboratory includes laboratories for radioanalysis and low-level detection and measurement of radioisotopes. Radioactive sources are also manufactured in this laboratory.

The experiments or processes used in the radiochemical portion of the 329 Laboratory include dissolution of solids, ion-exchange and precipitation separations, and liquid extractions. A description of the waste typically generated in the radiochemistry portion of the 329 Laboratory follows:

- Volume--1 kgal/yr
- Chemical Composition--nitrate, carbonate, chlorine, oxalate, sulfate, fluorine, sodium, and ammonia

- Predominant Radionuclides--americium-241 ( $^{241}\text{Am}$ ),  $^{60}\text{Co}$ ,  $^{137}\text{Cs}$ , iron-55 ( $^{55}\text{Fe}$ ), niobium-93m ( $^{93\text{m}}\text{Nb}$ ), nickel-63 ( $^{63}\text{Ni}$ ), plutonium-239 ( $^{239}\text{Pu}$ ) and -240 ( $^{240}\text{Pu}$ ), and  $^{90}\text{Sr}$ .

Only a small amount of waste is produced in the low-level detection facility. A general description of the waste produced follows:

- Volume--0.01 kgal/yr
- Chemical Composition--water
- Predominant Radionuclides-- $^{60}\text{Co}$ ,  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ .

### 2.1.6 3720 Building

Several laboratories are housed in the 3720 Building. Of these only the Geochemistry group currently generates radioactive waste as a result of the study of radioactive grouts and their leachates. The small amount of radioactive waste generated in the 3720 Building is collected in barrels and transported to the 340 Facility where it is added to the accumulation tanks. A general description of the waste being generated follows:

- Volume--0.2 kgal/yr through fiscal year (FY) 1990
- Chemical Composition--varies depending on experiment
- Predominant Radionuclides--iodine-125 ( $^{125}\text{I}$ ) and technetium-99 ( $^{99}\text{Tc}$ ).

The other project currently being conducted in the 3720 Building that generates tank wastes is the result of field lysimeter studies. A general description of the lysimeter waste that will be generated from lysimeter studies follows:

- Volume--0.1 kgal/yr decreasing by 25%/yr
- Chemical Composition--varies depending on experiment
- Predominant Radionuclides--none detected.

### 2.1.7 331 Life Sciences Laboratory

The 331 Life Sciences Laboratory is used for a variety of biological and ecological research studies. A small amount of tank waste is generated at the 331 Laboratory from the various research projects. The waste is transported to the 340 Facility in drums and added to the accumulation tanks through the decontamination sump. A general description of the waste follows:

- Volume--0.1 kgal/yr

- Chemical Composition--dilute nitric acid
- Predominant Radionuclides-- $^{14}\text{C}$ , tritium ( $^3\text{H}$ ), and  $^{125}\text{I}$ .

### 2.1.8 340 Facility

The 340 Facility is the terminus of the RLWS servicing the 300 Area. The RLWS is an encased stainless steel piping system that connects the 324, 325, 326, 327, and 329 Buildings to the 340 Facility storage tanks. The 340 Facility has two 15,000-gal-capacity vault storage tanks, of which one is always on line; in addition, should the capacity of the vault tanks be exceeded, six 8,000-gal-capacity above-ground storage tanks would be used to hold liquid. The 340 Facility, in addition to acting as an accumulation and storage facility, also has facilities for decontamination, miscellaneous storage, and pumping to rail tank cars.

In addition to direct use of the RLWS, radioactive wastes enter the 340 Facility through the addition of containerized waste trucked to the 340 and cross connections of the retention process sewer (RPS) to the RLWS.

The RPS system is used to send nonhazardous waste, which has the potential to become contaminated, to the 300 Area Process Trench. Before the waste is allowed to reach the trench, it passes through at least two radiation monitoring detectors (diverter stations). If higher-than-set-point radiation levels are detected, the RPS waste is diverted into the RLWS piping through a tie-in-leg.

The average composition of the waste handled by the 340 Facility during FY 1989 is provided in Table A.2-1.

Table A.2-1. The Average Composition of the Waste Handled in the 340 Facility during Fiscal Year 1989.

Constituent	Concentration
Activity	
Total Alpha	5.4 E+01 $\mu$ Ci/gal
Total Beta	5.7 E+03 $\mu$ Ci/gal
Fissile	
$^{239/240}\text{Pu}$	4.9 E-05 g/L
$^{235}\text{U}$	5.7 E-05 g/L
Major MFP Contributors	
$^{134}\text{Cs}$	4.8 E+01 $\mu$ Ci/gal
$^{137}\text{Cs}$	5.5 E+02 $\mu$ Ci/gal
$^{144}\text{Ce}$	1.9 E+02 $\mu$ Ci/gal
$^{106}\text{Ru}$	3.5 E+01 $\mu$ Ci/gal
$^{90}\text{Sr}$	7.8 E+01 $\mu$ Ci/gal
Chemical Composition and Properties	
pH	8.37
OH	6.5 E-06 mol/L
Cl	2.6 E-03 mol/L
$\text{PO}_4$	2,519 ppm
$\text{NO}_2$	79 ppm
$\text{NO}_3$	817 ppm
Al	1,200 ppm
specific gravity	1.0178
solids (%)	0.02

ppm = parts per million

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### 3.0 NEW WASTE GENERATORS AT THE 400 AREA

#### 3.1 DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED

The 400 Area contains the Fast Flux Test Facility (FFTF), a U.S. Government-owned nuclear reactor plant specifically designed for the irradiation and testing of nuclear reactor fuels and materials. The FFTF has played a key role in developing and testing fuels and materials for application in fast neutron flux reactors and in testing of fusion reactor materials.

The 400-MW fast-breeder reactor is located in a shielded cell in the center of the containment building. The heat generated by the fission process is removed from the reactor by liquid sodium circulating under low pressure through three primary coolant loops. An intermediate heat exchanger in each of these three loops separates the radioactive sodium in the primary system from the nonradioactive sodium in the secondary system. The radioactive primary sodium does not leave the Reactor Containment Building. Three secondary sodium loops transport reactor heat from the intermediate heat exchangers to the air-cooled tubes of the 12 heat dump exchangers.

The FFTF also includes facilities for receiving, conditioning, storing, and installing core components and test assemblies. Examination and packaging capabilities for onsite and offsite shipments and radioactive waste handling are provided.

#### 3.2 GENERATION OF TANK WASTES IN THE 400 AREA

In the 400 Area, radioactive liquid wastes are generated primarily in conjunction with the removal of residual sodium from irradiated reactor components and fuel assemblies in the Interim Examination and Maintenance (IEM) Cell and by the cleaning and decontamination activities conducted in the Maintenance and Storage Facility (MASF). Wastewater, generated during the cleaning processes, is stored in a 5,000-gal-capacity tank at the FFTF and two 5,000-gal-capacity tanks at the MASF. The wastewater is moved from the FFTF to the MASF via an 8,000-gal-capacity rail car and then transferred to the 200 Area tank farms via a 20,000-gal-capacity rail tank car. Shipments of the contaminated wastewater to the 200 Area tank farms occur approximately once a year.

#### 3.3 TANK WASTE MINIMIZATION AT THE FAST FLUX TEST FACILITY AND MAINTENANCE AND STORAGE FACILITY

The design of the cleaning systems used in the IEM cell is such that the washwater is recirculated to the greatest extent possible, minimizing the amount of radioactive tank waste generated by the facility. Current practices generate about 500 gal of contaminated water per cleaning evolution. The total quantity of wastewater generated in the IEM cell is dependent on the number of reactor assemblies washed in that year.

The amount of wastewater generated by the IEM cell and MASF annually is less than that required to perform the annual hydrological test on the 8,000-gal-capacity tank car used to ship the waste to the tank farm. To further minimize the amount of tank waste generated in the 400 Area, procedures have been changed to allow use of the existing wastewater to fill the tank car for the required annual hydrological testing, resulting in a substantial reduction in the volume of wastewater generated annually.

#### 3.4 FUTURE TANK WASTE GENERATED AS A RESULT OF THE FAST FLUX TEST FACILITY SHUTDOWN OPTION

The future of FFTF and MASF is undetermined at this time. If the reactor is shut down permanently, the amount of wastewater generated would depend upon the type of sodium disposal system used. The possibility exists for the generation of up to 500,000 gal of radioactive 50% sodium hydroxide solution, produced from the reaction and disposal of the sodium drained from the FFTF cooling systems. If this material cannot be used for neutralization at the Plutonium-Uranium Extraction (PUREX) Plant, it will have to be treated as radioactive waste (probably by evaporating off the water and converting of hydroxide to sodium carbonate). In addition, an additional 250,000 gal of slightly contaminated and low-level radioactive water or alcohol may be generated as a result of sodium removal operations in FFTF piping and components after the sodium systems are drained.



## 4.0 TANK FARMS

### 4.1 DESCRIPTION OF THE FACILITIES AND TYPES OF WASTE GENERATED

#### 4.1.1 Single-Shell Tanks

Between 1943 and 1964, 149 single-shell tanks (SST) were built in the 200 East and 200 West Areas of the Hanford Site for storage of radioactive wastes. These SSTs are located in 12 tank farms, with each tank farm consisting of 4 to 18 SSTs.

The SSTs have volumes of 55,000 to 1,000,000 gal. One hundred thirty-three of the SSTs are 75 ft in diameter and 29.75 to 54 ft high (at their highest point) with nominal capacities of 500,000 to 1,000,000 gal. Sixteen of the SSTs are smaller units of similar design, 20 ft in diameter and 25.5 ft high with capacities of 55,000 gal each.

The tanks are located below grade with at least 6 ft of soil covering the tanks to provide shielding and minimize the radiation exposure to tank farm operating personnel. Most of the 500,000- and 750,000-gal-capacity SSTs were built in the form of "cascades" of three or four SSTs each. Waste was transferred to the first SST in the cascade and allowed to overflow into each of the successive SSTs in the cascade through inlet and overflow lines located near the top of the steel liner provided in each SST.

Access to each of the SSTs is provided by risers penetrating the domed top of the SSTs. These risers vary in diameter from 4 to 42 in. Each of the SSTs have up to 11 risers with the majority of the SSTs having 3 to 5 risers.

Radioactive waste generated during the various Hanford Site operations was not placed into SSTs after November 1980. While the SSTs are considered to have been "taken out of service" in November 1980, the 149 tanks continue to hold approximately 37 Mgal of saltcake, sludge, and interstitial liquid.

#### 4.1.2 Double-Shell Tanks

Between 1968 and 1986, 28 double-shell tanks (DST) were constructed: 3 of these tanks are located in the 200 West Area (241-SY Farm) and an additional 25 tanks are located in the 200 East Area (241-AN, -AP, -AW, -AY, and -AZ Tank Farms). All of these DSTs were constructed at least 5 ft below grade to provide shielding to minimize the radiation exposures of operating personnel. Table A.4-1 provides a chronology of the DST construction.

The four 241-AY and -AZ tanks each have a 1-Mgal capacity and are designed to store the high-heat-generating neutralized current acid waste (NCAW) from the PUREX process. These tanks are referred to as aging waste tanks and have air-lift circulators for mixing and a vessel ventilation system designed to remove and condense steam.

Table A.4-1. Chronology of the Double Shell-Tank Construction.

Tank Farm	Built	Tank quantity	Tank volume (Mgal)	Comment
241-AY	1968-70	2	1.000	Aging waste tank
241-AZ	1971-77	2	1.000	Aging waste tank
241-SY	1974-76	3	1.140	
241-AW	1978-80	6	1.140	
241-AN	1980-81	7	1.140	
241-AP	1983-86	8	1.140	

The DSTs use a tank-within-a-tank design to provide double containment of the radioactive liquid and solid wastes they contain. This design ensures that, in the event of a leak in the primary shell, the liquid waste will be fully contained by the outer shell.

The freestanding primary tank is about 75 ft in diameter and 46 ft high at the dome crown. The carbon steel in the bottom of the tank ranges from 1/2 to 1 in. in thickness. The primary tank wall thickness ranges from 1/2 to 3/4 in. with the dome thickness at 3/8 in.

An annular space of 2.5 ft is provided between the primary tank and the secondary steel tank to allow for installation of liquid-level and leak detection devices; inspection equipment such as periscopes, television cameras, and photographic cameras; ventilation air supply and exhaust ducts; and equipment for pumping liquid out of the annular space.

Sixty-four tank dome penetrations in the primary tank and annulus allow for various monitoring and processing activities. Primary tank monitoring activities include measurement of liquid level, sludge level, temperature, and pressure.

#### 4.1.3 Solutions Added to Double-Shell Tanks by Tank Farm Operation

The tank farm facilities at the Hanford Site receive radioactive wastes generated by other Hanford Site waste generators. Tank farm operations are typically characterized as a waste receiver rather than a waste generator. However, in the operation of the tank farms, a variety of flushes and chemical additions are made that increase the volume of the wastes in the tanks. These streams are identified because their minimization has the overall effect of reducing the volume requiring treatment for final disposal.

1. Salt Well Liquor--The SSTs contain moist solids (salts and sludges) that contain interstitial liquid. Saltwell pumping can remove a portion of the interstitial liquid called salt well liquor (SWL) from these solids. Before October 1989, 101 SSTs had been pumped, leaving 48 SSTs to be pumped by the end of FY 1995 (Tri-Party Agreement Milestone M-05).

It is estimated that 901,000 gal of pumpable liquid will be removed from SSTs and transferred to DSTs between April 1990 and June 1991. It is predicted that 4,000,000 gal will be removed from the SSTs by FY 1995 when the saltwell pumping program is expected to be completed.

2. Air-Lift Circulator (ALC) Flushes--Salts are periodically flushed from the ALC in the aging waste tanks. The estimated volume of ALC water flushes between April 1990 and June 1991 is 114,400 gal.
3. Aging Waste Ventilation System Condensate and De-entrainer Flushes--The radionuclide concentration in these two dilute streams is too high to be sent to cribs, so these solutions are returned to the DSTs. This is expected to add an estimated 199,000 gal of waste to the DSTs between April 1990 and June 1991.
4. Caustic Addition to DST 241-AN-107--Slow chemical reactions in DST 241-AN-107 consume hydroxide ion. To maintain the proper pH in the DST, caustic is added to replenish the hydroxide ion concentration. An estimated 32,500 gal are expected to be added to the DST volume between April 1990 and June 1991.
5. Steam Condensate from the Tank Farm 241-SY Ventilation System--Tank Farm 241-SY contains three DSTs in the 200 West Area (physically separated by 5 to 6 mi from 25 DSTs in the 200 East Area.) The volume of Tank Farm 241-SY steam condensate is estimated to be 6,600 gal between April 1990 and June 1991.
6. Washdown of Measurement Equipment (e.g., level indicators, leak detection pits)--Measurement equipment that contacts tank waste periodically must be washed to remove accumulated solids and salts. Occasionally, solution must be removed from leak detection pits. The estimated volume to removed between April 1990 and June 1991 is 2,800 gal.
7. DST 241-AZ-101 Aging Waste Steam Condensate--The DST 241-AZ-101 contains steam coils to boil water from the aging waste. To prevent these steam coils from freezing during winter weather, a small amount of steam must be allowed through the coils which will produce an estimated 4,700 gal of condensate between April 1990 and June 1991. The steam condensate is no longer allowed into Crib 216-A-08 because of the listed waste issue.
8. Tank Car Waste Flushing and Water from Recertification--Radioactive waste is shipped by rail tank car to the 200 East DSTs from the 100-N, 300, and 400 Areas. The tank car used to transport this waste must be flushed and recertified. The estimated volume of waste to be generated during these operations between April 1990 and June 1991 is 99,000 gal.
9. Miscellaneous Wastes (e.g., Evaporator 242-S Steam Leaks, pump room sumps, RC-1 Sampler)--The volume of miscellaneous solutions generated between April 1990 and June 1991 is estimated to be 37,500 gal.

## 4.2 WASTE MINIMIZATION ACTIVITIES

### 4.2.1 Past Waste Minimization Activities

Since 1944 about 263,000,000 gal of tank space has been reclaimed. Most of the treatment consisted of evaporating the waste to remove water; however, there has also been some reprocessing of the wastes to remove specific isotopes. During the 1950s tank wastes were reprocessed to recover uranium. During the late 1960s and 1970s tank wastes were again reprocessed to recover cesium and strontium.

The evaporators used to reduce tank waste volumes were operated through the following dates (dates are approximate within 1 yr):

- Evaporator 242-B (1952 to 1955)
- Evaporator 242-T (1952 to 1976)
- In-Tank Solidification Unit 1 (1965 to 1974)
- In-Tank Solidification Unit 2 (1968 to 1974)
- REDOX Concentrator (1967 to 1972)
- B Plant Concentrator (1967 to 1968)
- Evaporator-Crystallizer 242-S (1973 to 1980)
- Evaporator-Crystallizer 242-A (1976 to present).

### 4.2.2 Present Waste Minimization Activities

Forecasts that current rates of waste generation will fill the DSTs in 1991 have prompted a Hanford Site-wide effort to significantly reduce the amount of waste sent to the DSTs. Within the tank farm operating area the following waste-avoiding activities have been adopted.

1. The frequency of the Ventilation System 702-A de-entrainer flush has been reduced, thus avoiding 61,500 gal waste between April 1990 and June 1991. This reduced the de-entrainer flush volume from 260,500 to 199,000 gal between April 1990 and June 1991.
2. Tank Farm 241-AZ air-lift circulator flush was reduced by 50%, thus avoiding 78,000 gal of waste generation between April 1990 and June 1991.
3. The frequency of Catch Tank 152-AX water jet transfer was reduced, thus avoiding 57,000 gal of waste between April 1990 and June 1991.
4. A process test in DST 241-AZ-102 was cancelled, thus avoiding the generation of 30,000 gal of steam condensate from the heater coils.
5. The flushing of the 241-AY Tank Farm air-lift circulators have been cancelled, thus avoiding 45,000 gal of water added to the DSTs between April 1990 and June 1991.
6. Several miscellaneous streams have been eliminated, thus avoiding the generation of 131,000 gal of waste between April 1990 and June 1991.

#### 4.3 SUMMARY

Waste avoidance activities within Tank Farm Operations are expected to reduce the volume of waste sent to the tanks by 400,000 gal between April 1990 and June 1991. For this period, the forecast for waste generated within Tank Farms has been reduced from 1,800,000 to 1,400,000 gal.

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## 5.0 EVAPORATORS

### 5.1 DESCRIPTION OF EVAPORATOR FACILITIES

Since the early 1950s, eight evaporator facilities have been used to treat tank wastes at the Hanford Site. The only evaporator facility that is planned for continued operation is Evaporator-Crystallizer 242-A located in the 200 East Area.

Evaporator-Crystallizer 242-A is used to reduce the volume of waste requiring treatment for disposal. The evaporator operates under a vacuum and employs evaporative concentration. When the concentrate is pumped to DSTs and cools, salt crystals precipitate.

### 5.2 TYPES OF WASTE GENERATED

The operation of the Evaporator-Crystallizer 242-A does not generate new tank waste except when there is a process upset. The following streams are generated:

- DSS, which is returned to DSTs
- Steam condensate from reboiler, which is sent to the 216-B-3 Pond
- Process condensate, which is held for treatment
- Cooling water from the process condenser, which is sent to the 216-B-3 Pond
- Small-volume, intermittent wastes such as de-entrainer wash, which are sent to the evaporator pot.

The slurry returned to the DSTs was originally a DST waste before being pumped into the evaporator, so it is not considered an original waste stream for the tank farms.

If there is an upset condition and process condensate becomes contaminated with radionuclides, the process condensate may be returned to a DST. This seldom occurs and the process condensate is typically not considered a tank waste.

Previously, the process condensate was discharged untreated to the Hanford Site soil column in the 200 East Area. This practice has been discontinued and a new collection, treatment, and processing facility is being constructed.

The small-volume, intermittent wastes such as de-entrainer wash, are sent to the evaporator pot where their identity is lost during evaporation with DSS.

### 5.3 WASTE TREATMENT ACTIVITIES

#### 5.3.1 Past Waste Treatment Activities

Since 1944 about 263,000,000 gal of tank space has been reclaimed. Most of this space was reclaimed through evaporation of the waste to remove water.

Evaporator-Crystallizer 242-A began operation in 1976 and has evaporated more than 65,000,000 gal of water from wastes stored in various tanks. This is approximately 25% of all tank waste volume reduction achieved at the Hanford Site.

#### 5.3.2 Present Waste Treatment Activities

Evaporator-Crystallizer 242-A will resume operation after improvements and additions are completed in 1991. Operation of the evaporator will reduce the volume of liquids stored in DSTs.



## 6.0 PLUTONIUM FINISHING PLANT

### 6.1 PLUTONIUM FINISHING PLANT FACILITY DESCRIPTION

The Plutonium Finishing Plant (PFP) is located in the 200 West Area of the Hanford Site. The PFP has the primary mission of plutonium processing, handling, and storage. Plutonium metal production and reclamation, waste treatment, product storage, and packaging for shipment are the principle operations conducted at the PFP.

The PFP houses the Remote Mechanical C (RMC) line which converts plutonium nitrate solution to plutonium metal, using a process of precipitating plutonium nitrate with oxalic acid to form plutonium oxalate, filtering the plutonium oxalate and calcinating it to produce plutonium oxide. The plutonium oxide is fluorinated by contact with gaseous hydrofluoric acid and oxygen to produce plutonium fluoride. This plutonium fluoride is reduced in the presence of calcium metal to produce plutonium metal.

Building 236-Z houses the Plutonium Reclamation Facility (PRF), which produces plutonium nitrate from recovered plutonium scrap solutions and solids. Additionally, PRF is used for slag and crucible dissolution and processing filtrate from the RMC line.

Building 241-Z acts as intermediate storage for liquid radioactive and chemical wastes from the RMC and PRF production processes and laboratory waste. These wastes are eventually transferred to the tank farms.

### 6.2 WASTE TYPE DESCRIPTION

The major waste stream sources are the active RMC line and the PRF. High-salt and low-salt wastes, stored in Building 241-Z, result from the RMC line, PRF, and the laboratories.

### 6.3 WASTE MINIMIZATION ACTIVITIES

#### 6.3.1 Implementation of TRUEX at the Plutonium Finishing Plant

Past waste minimization activities at PFP evaluated the benefits of the transuranic-extraction (TRUEX) process and its implementation at the PFP. This evaluation concluded that the TRUEX process should be implemented at the PFP to recover plutonium that was being discarded as waste.

Changes in the defense production mission at the Hanford Site have encouraged a re-evaluation of the TRUEX process implementation. Implementation of the TRUEX process remains a recommended addition to PFP because it will provide several benefits:

- Reduce the cost of final waste disposal
- Recover plutonium otherwise lost as waste
- Enhance PFP for future missions
- Minimize hazardous chemical wastes.

### 6.3.2 Hanford Private Sector Participation Conference

A recent conference sponsored by the U.S. Department of Energy-Richland Operations Office (DOE-RL) requested proposals for processing PFP liquid wastes into a solid TRU waste form for disposal at the Waste Isolation Pilot Plant (WIPP) and a solid low-level waste (LLW) for disposal at the Hanford Site LLW burial grounds. Technical information was provided to the private sector to permit development and design of processes for liquid-to-solid waste conversion.

### 6.3.3 Plutonium Reclamation Facility Process Modification

In the PRF process, feed solutions are extracted in the CA Column with an organic solution and the resulting plutonium-rich organic solution is stripped in the CC Column to produce plutonium solution. The current feed point valve for the CC Column is approximately 10 ft below the top of the column, allowing efficient use of approximately 200 contact plates. A proposed modification to the process would use the CC Column feed valve at the top of the column, permitting use of all 270 contact plates. In FY 1988, the PRF process was modified to determine the actual benefits of using the entire CC Column of contact plates. The FY 1988 modified PRF process campaign resulted in several advantages:

- Reduced metallic impurities
- Decreased waste generation
- Improved plutonium recovery
- Reduced process upset recovery time by up to 60%.

Another modification to the PRF process resulting from waste minimization activities is the proposal to bypass the OA Column during uranium depletion. The OA Column is used only during plutonium-uranium partitioning; therefore, it may be bypassed during plutonium-only and uranium depletion operations. Resulting benefits of this modification are listed below:

- Waste reduction approaching 3,000 gal/mo
- Reduction of corrosion in Tanks 37 and 38
- Concurrent operation of the PRF and RMC line
- Elimination of pump use.

#### 6.3.4 Segregation of Waste at the RMC Line

A proposed process change to the RMC line aimed at minimizing waste and increasing ease of plutonium-bearing material reprocessing, is currently being evaluated. This proposed change would segregate the different components of the RMC line crucible waste before reprocessing at PRF, allowing only plutonium-bearing material to be treated. The segregated solid waste will be disposed of in solid form. Production of DST waste would be reduced.

### 6.4 PLUTONIUM RECLAMATION FACILITY WASTE MINIMIZATION PLAN

#### 6.4.1 Purpose

A waste minimization plan for PFP was developed in FY 1989 to provide guidance on minimizing the production of hazardous wastes in an economically feasible manner and consistent with safe plant operations. Areas addressed in the plan include organizational responsibilities, training, employee participation and incentive program, and incorporation of waste minimization as part of the design process for new projects or designs.

#### 6.4.2 Employee Training

All employees of PFP are scheduled for training in a "Hazardous Materials/Waste-Facility Specific" class to start them thinking of waste minimization as an everyday achievable activity. The class will provide the employees with a definition of waste minimization, an overview of the program at PFP, examples of waste minimization proposals and their role in the waste minimization effort.

#### 6.4.3 New Projects and Designs

New projects and designs will be required to include waste minimization as an integral part of the design process. Normally, this effort will be in the conceptual design stage of the project. Design review committees will include waste minimization as part of their task in reviewing a new project or design.

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## 7.0 PUREX PLANT

### 7.1 DESCRIPTION OF THE FACILITY

Construction of the PUREX Plant in the Hanford Site 200 East Area was initiated in 1952 by the U.S. Atomic Energy Commission (now the U.S. Department of Energy [DOE]). The plant processes irradiated nuclear reactor fuels for the recovery of uranium and plutonium.

The nuclear material processing in the PUREX Plant takes place in three parts (see Figure A.7-1):

1. **Head-end**--Irradiated fuel elements are chemically declad and the fuel is separately dissolved.
2. **Solvent-extraction**--Chemical processes extract and separate plutonium, neptunium, and uranium from the dissolved fuel. Plutonium nitrate liquid can either be transported to the PFP for conversion to plutonium metal or transferred to the plutonium oxide production facility within the PUREX Plant.
3. **Plutonium Oxide Production**--The plutonium nitrate liquid is converted into plutonium oxide powder. The powder is packaged, weighed, and prepared for shipment.

Liquid wastes, containing MFPs, are neutralized at the PUREX Plant and routed to the tank farms for storage.

### 7.2 TYPES OF WASTE GENERATED

The PUREX Plant generates two major wastes: NCAW and neutralized cladding removal waste (NCRW). The NCAW is the aqueous high-salt waste from the first-cycle solvent extraction column in the PUREX Plant. This waste is neutralized to prevent corrosion of the carbon steel tanks. Cladding removal waste (CRW) results from the dissolution of the N Reactor spent fuel Zircaloy cladding using the Zirflex process in the PUREX Plant.

### 7.3 WASTE MINIMIZATION

#### 7.3.1 Past Waste Minimization and Pretreatment Activities

In previous years continuing efforts have been made in the area of waste minimization:

- The removal of cesium and strontium from the high-level wastes (HLW) (activity completed).
- The denitration of HLW, thereby reducing the amount of caustic required for neutralization and thus the total amount of the wastes. A flowsheet is given in Figure A.7-1 showing the PUREX Plant product and waste streams.

- The recycling of process condensates to minimize the amount requiring disposal in the cribs.
- The use of improved diluent (normal paraffin hydrocarbon [NPH]) enabled eliminating the sodium hydroxide treatment of the first-cycle solvent which significantly reduced the amount of organic wash waste (OWW).
- The improved diluent also enabled reducing the changeout frequency of sodium carbonate-potassium permanganate solvent wash solutions, thus further reducing the quantity of OWW.
- A number of flowsheet and equipment changes have resulted directly in reduced waste volumes or in improved product quality, thus reducing the amount of rework and associated waste volume. These include replacing the plutonium anion exchange system with the third plutonium cycle solvent extraction system, eliminating the use of the HS (T-H3) column which reduced the volume of scrub solution added, reducing the concentration of potassium permanganate in the solvent wash solution, and eliminating the addition of sodium nitrite to the backcycle waste feed tank (TK-F10).

### 7.3.2 Present Waste Minimization and Pretreatment Activities

The following waste reduction schemes are in the development stages:

- The use of rare earth precipitation process to reduce much of the TRU material in decladding waste. A flowsheet is given in Figure A.7-2.
- The destruction of the ammonia resulting from fuel decladding and the recycle of ammonia scrubber distillate to reduce the amount of liquid waste. A flowsheet, before and after the ammonia destruction installation, is given in Figures A.7-3 and A.7-4.

### 7.4 WASTE MINIMIZATION AND PRETREATMENT PLAN SUMMARY

Substantial reductions in the amounts of wastes discharged from the PUREX Plant have been made from past practices, and a number of activities for further reductions are being evaluated for possible future application.

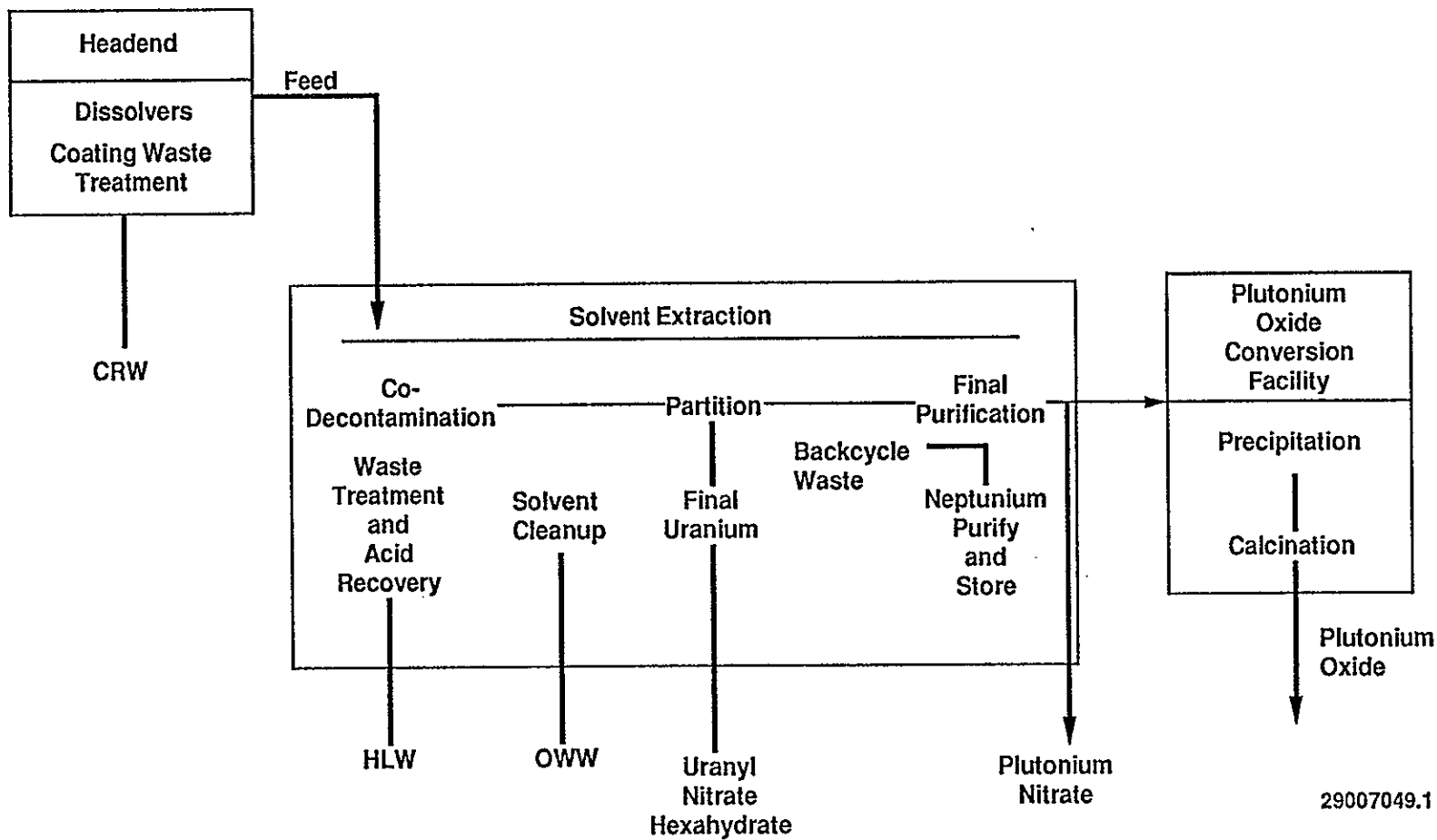


Figure A.7-1. Simplified PUREX Plant Flowsheet.

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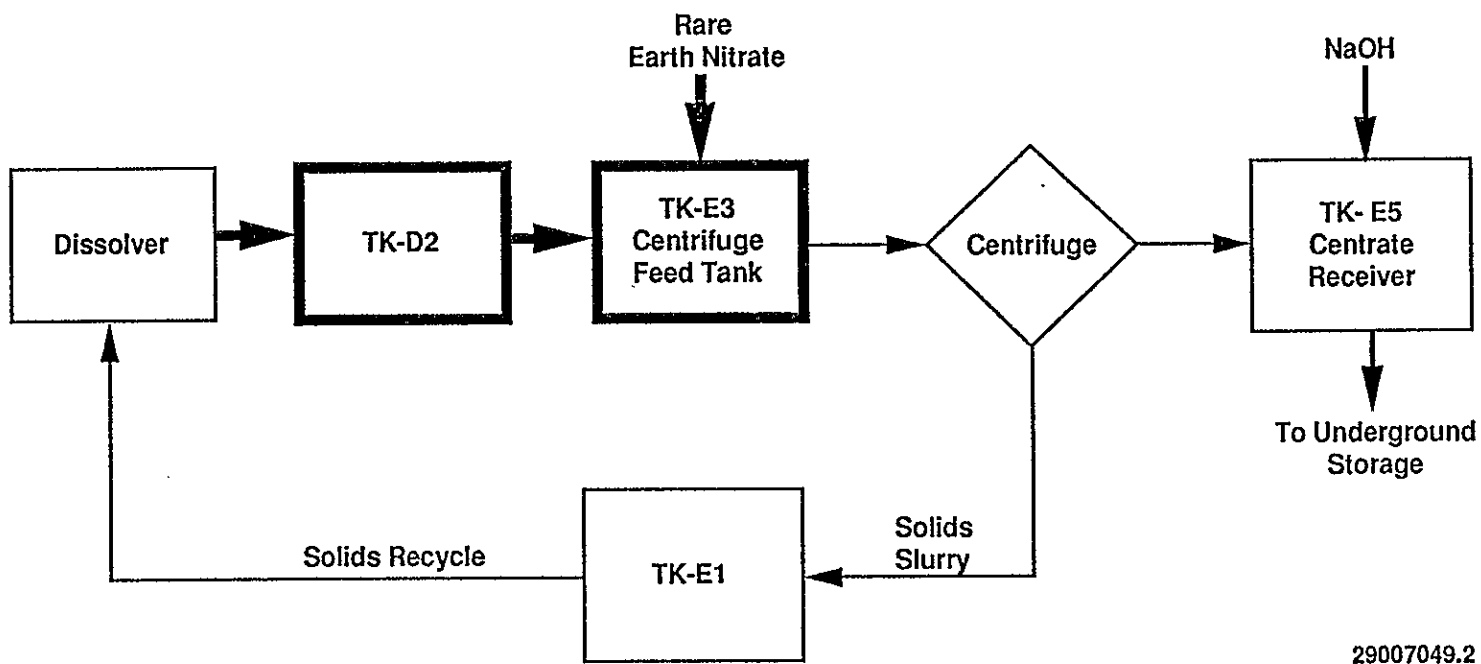
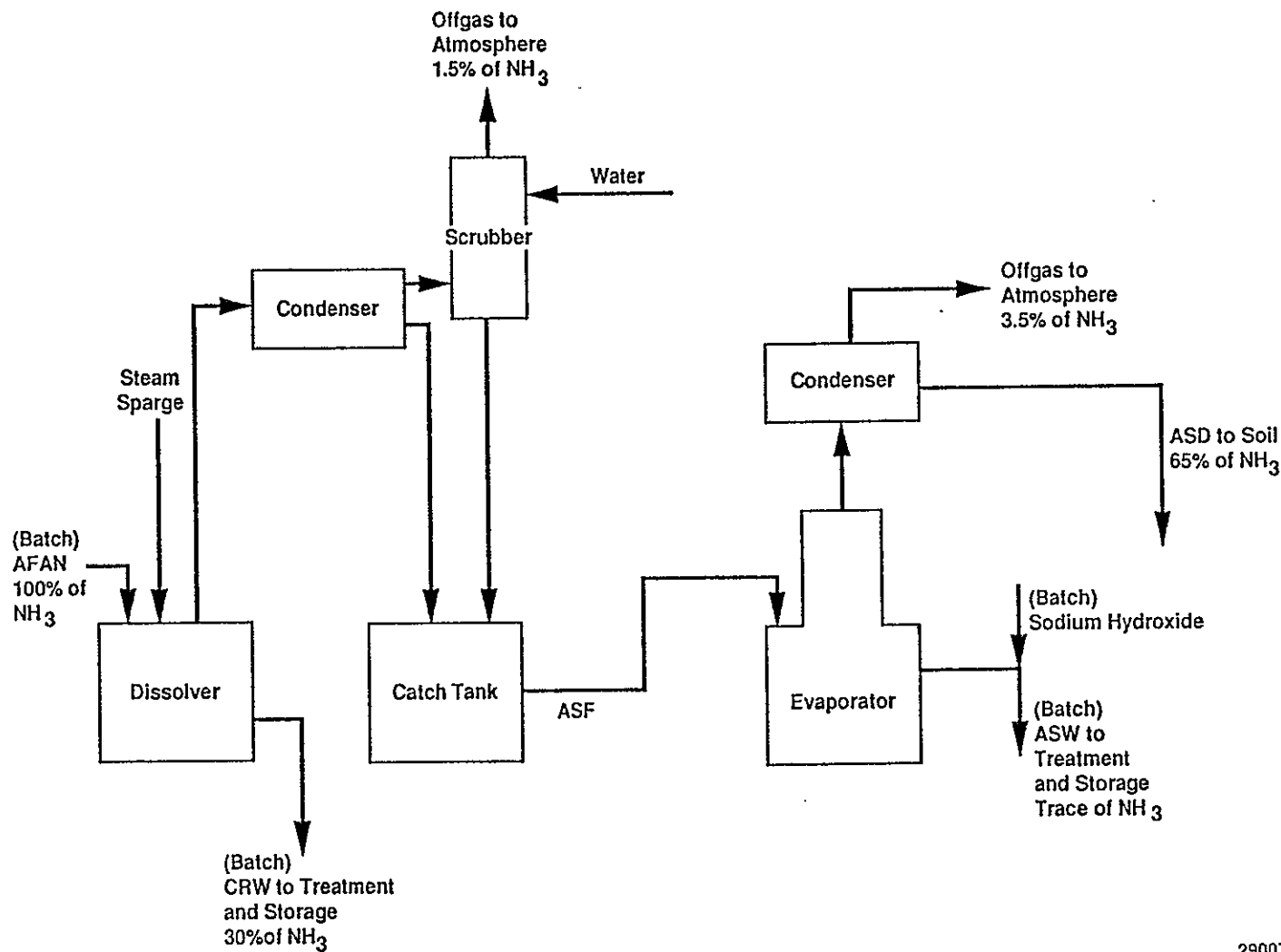


Figure A.7-2. Rare Earth Precipitation Process.

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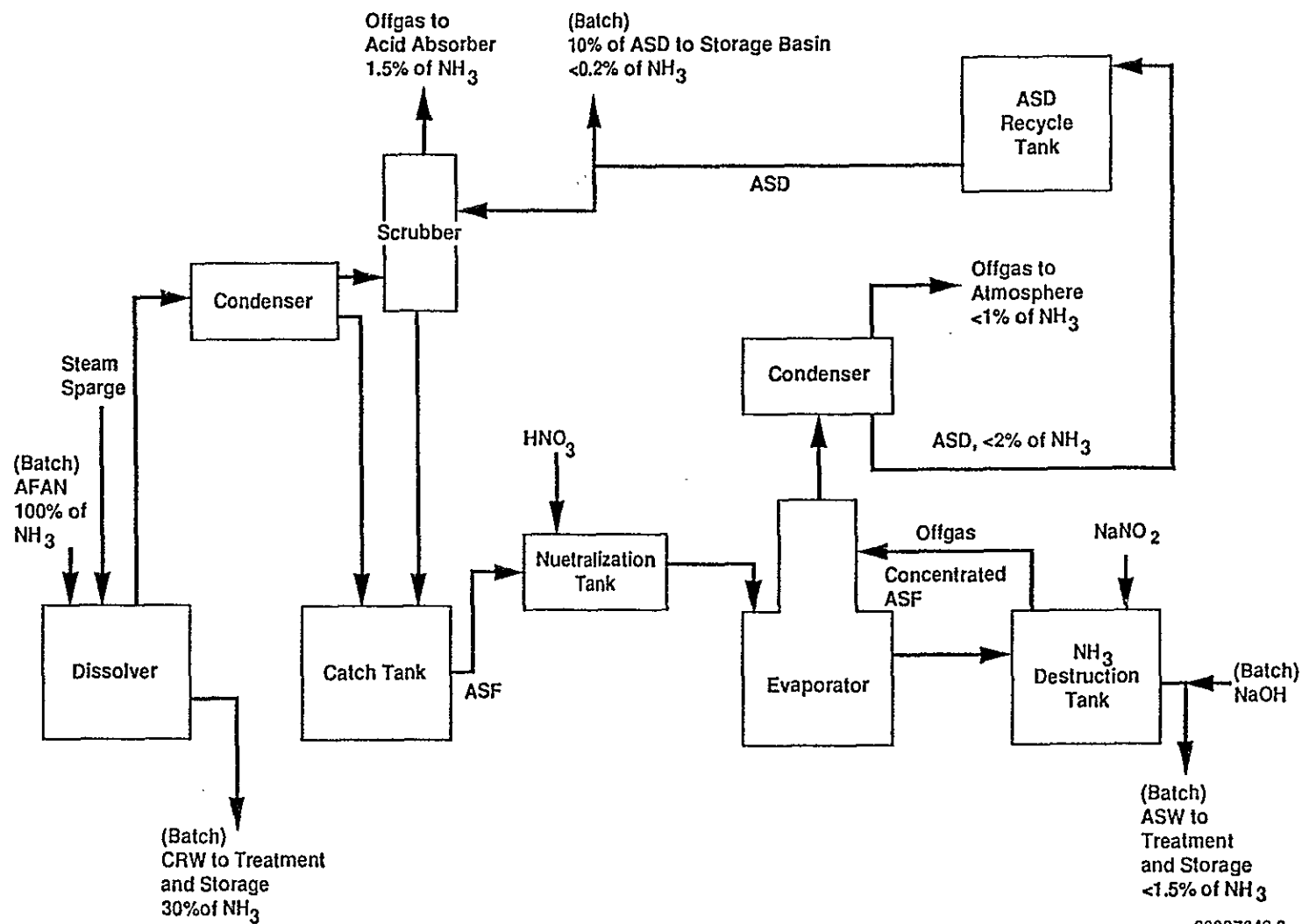
Figure A.7-3. PUREX Plant Ammonia Waste Handling--Before Ammonia Destruction.



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Figure A.7-4. PUREX Plant Ammonia Waste Handling--With Ammonia Destruction.



## 8.0 B PLANT

### 8.1 DESCRIPTION OF B PLANT AND TYPE OF WASTES GENERATED

#### 8.1.1 Description of Facility

B Plant is a structure more than 800 ft long and 72 ft high, with concrete walls a minimum of 4 ft thick. It is designed to remotely process radioactive materials with no radiation exposure to operators. The first mission of B Plant was to reprocess spent fuel between 1945 and 1952 using the bismuth phosphate process.

B Plant was refurbished for Mission 2 (1965 to 1985) to recover and purify cesium and strontium from newly generated current acid waste (CAW) and from stored wastes in tanks (NCAW). The facility is now being refurbished for Mission 3 to pretreat tank wastes before vitrification in the Hanford Waste Vitrification Plant (HWVP).

#### 8.1.2 Types of Waste Generated at B Plant

Presently B Plant produces about 1 Mgal/yr of contaminated water from a miscellaneous assortment of activities (e.g., flushing, cleaning) required to maintain the building in a functional state.

### 8.2 WASTE TREATMENT ACTIVITIES

#### 8.2.1 Past Waste Treatment Activities and Process Flowsheets

8.2.1.1 Mission 1 (1945 to 1952). In June 1949 the concept of concentrating first decontamination cycle wastes by evaporation was proposed. After determining that the approach was economically feasible, Evaporator 242-B was built and operated from 1951 to 1954.

More than 6 Mgal of first-cycle waste was processed achieving a volume reduction of 81%. Then an additional 2.3 Mgal of space were recovered from uranium recovery wastes.

8.2.1.2 Mission 2 (1965 to 1985). B Plant was refurbished to remove cesium and strontium from self-boiling wastes stored in tanks or generated by the PUREX process. The high radiation of cesium and strontium was responsible for the heat and elevated radiolytic decomposition of water in the wastes.

With cesium and strontium removed, the heat generation and radiolytic decomposition was significantly reduced such that the residual waste could be solidified and stored more safely. In effect, the 1965 to 1985 campaign was a massive waste treatment effort that resulted in improved waste storage at the Hanford Site.

**8.2.1.3 Preparation for Mission 3 (1985 to 1990).** Vessel cleanout and routine maintenance activities between 1985 and 1990 have generated dilute liquid wastes. The volume of waste sent to tank farms was reduced in the early part of this period by concentration in the B Plant concentrator.

The water evaporated from the B Plant concentrator (process condensate) had been discharged to Crib 216-B-62. A change in discharge criteria resulted in the cessation of discharge to Crib 216-B-62 and influenced the development of the B Plant Process Condensate Treatment Facility.

Since the discharge to Crib 216-B-62 was stopped, dilute liquid wastes generated in B Plant have been sent directly to the tank farms without concentration. When the B Plant Process Condensate Treatment Facility is constructed and becomes operational near the end of 1992, B Plant dilute waste will once again be treated by concentration before transfer to tank farms.

Approximately 1 Mgal/yr of dilute waste will be generated by B Plant and sent to tank farms until the startup of the B Plant Process Condensate Treatment Facility near the end of 1992.

## **8.2.2 Present Waste Treatment Activities and Process Flowsheets**

B Plant is presently being refurbished for its next mission. Activities are limited to cleanout and periodic testing of equipment.

## 9.0 S PLANT

### 9.1 DESCRIPTION OF FACILITIES AND TYPES OF WASTE

#### 9.1.1 Description of Facility

The S Plant (referred to on the Hanford Site as the REDOX Complex) area consists of the 202-S Canyon Building and the 222-S Analytical Laboratory. The 202-S Canyon Building was deactivated and is no longer producing tank wastes. The 222-S Analytical Laboratory conducts process support, environmental, and research and development activities.

The 222-S Laboratory is a dedicated laboratory facility. The laboratory currently provides analytical chemistry services in support of the Westinghouse Hanford Company (Westinghouse Hanford) processing plants. The laboratory work emphasizes waste management processing plants, environmental monitoring programs, B Plant, tank farms, Evaporator 242-A, the Waste Encapsulation Storage Facility (WESF), the PUREX Plant, the PFP, research support activities, and essential materials. The 222-S Laboratory is presently upgrading its facility, equipment, and procedures to support the environmental restoration program, SST and DST characterization programs, and the Resource Conservation and Recovery Act (RCRA) protocol programs. These additional missions may increase the 222-S Laboratory waste volume projections. Radioactive liquid waste at the 222-S Laboratory is generated by disposal of process and environmental samples and decontamination operations.

#### 9.1.2 Description of Waste

Most of the REDOX Complex waste is the result of the activities of the 222-S Laboratory. The waste stream consists of various neutralized acids.

Sodium hydroxide is added for neutralization and sodium nitrite is added to increase the nitrite concentrations to the tank farm specifications. The waste is agitated to ensure adequate mixing and is then transferred to tank farms for processing. After the transfer is complete, the lines are flushed with 1,000 gal of raw water. This waste is collected and blended with other 200 West Area wastes and transferred to the 200 East Area for treatment in Evaporator 242-A to produce DSSF for interim storage.

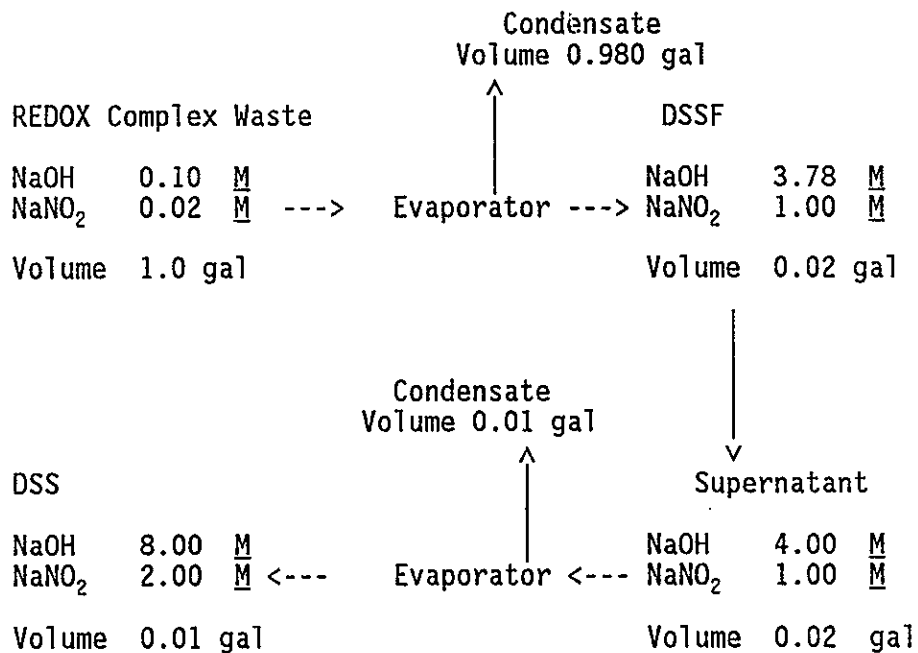
The intermediate-level waste streams contributing to Tank 101 are hood drains, decontamination hood number 16, hot laboratory sinks, and the inductively coupled plasma. The HLW streams contributing to Tank 103 are hot cell drains, slurping from decontamination hood number 16, 1-F manipulator repair hood drain, atomic absorption spectrophotometer drain, and hot tunnel sumps.

Table A.9-1 shows the composition of typical REDOX Complex wastes. Figure A.9-1 illustrates a flowsheet for the concentration of REDOX Complex waste. The volumes generated, chemical composition, radionuclide composition, and solids contents may vary according to the programs being supported by the laboratory.

Table A.9-1. REDOX Complex Waste Composition.

Chemical	Composition		
Liquids			
Carbonate	5.0 E-03	M	
Total organic carbon	1.0 E+00	g/L	
Fluoride	1.0 E-03	M	
Nitrite	2.5 E-02	M	
Nitrate	1.0 E-01	M	
Phosphate	5.0 E-03	M	
Sulfate	2.0 E-02	M	
Sodium	2.5 E-01	M	
Hydroxide	1.0 E-01	M	
Radionuclides			
Total alpha	5.0 E-06	Ci/L	
Total beta	2.0 E-04	Ci/L	
<sup>137</sup> Cs	5.0 E-05	Ci/L	
<sup>89/90</sup> Sr	3.0 E-05	Ci/L	
Plutonium	4.0 E-05	g/L	
Uranium	1.0 E-02	g/L	
Solids			
Percent	0.00 E+0		

Figure A.9-1. Concentration of REDOX Complex Waste.



The primary program currently being supported is SST characterization and the waste projections are based on the analysis schedule established for SST characterization.

## 9.2 WASTE MINIMIZATION

The projected waste generation volumes are based on the facility operating plans, the target waste generation rate and the SST and DST characterization schedules.

The projected waste generation volumes were baselined against the number of SST and DST core samples scheduled for analysis during FY 1990 at the 222-S Laboratory. The number of cores scheduled for analysis in FYs 1991 through 1994 is 10 cores/yr, increasing to 20 cores/yr in FYs 1995 through 2015. Volume projections will become more accurate as updated information is available.

Specific actions have been taken to minimize the volume of waste accumulated in the 219-S Facility and subsequently transferred to Tank 241-SY-102:

- Reduce flush volume to 500 gal of water per transfer.
- Reduce in half the flush volume after slurping samples.

Additional waste minimization activities are currently being evaluated for possible implementation.

The information presented in this section is based on two tank transfers analyzed at the 222-S Laboratory. Before this sampling, only hydroxide, nitrite, pH and total plutonium were analyzed before each transfer. Extensive chemical and radionuclide sampling will continue through FY 1990 and subsequent forecasts will be baselined against this data. The sampling and analyses will be performed according to RCRA protocol and will include organic solvents and extraction procedure determinations.



## 10.0 T PLANT

### 10.1 DESCRIPTION OF FACILITY AND TYPE OF WASTE GENERATED

#### 10.1.1 Description of Facility

The T Plant, located in the 200 West Area of the Hanford Site, has the primary mission of equipment decontamination and refurbishment. The T Plant waste system handles radioactive liquid waste from decontamination activities in the hot cells, the railroad tunnel, and the 2706-T Building. The railroad tunnel in Cell 2-L of T Plant generates waste from decontaminating railroad cars and multipurpose transfer boxes.

Most waste from cells in T Plant is generated by various decontamination processes. Each cell has a 6-in-diameter drain line that allows the waste to drain into the canyon 24-in-diameter sewer lines which empties into Tank 5-7, located in Cell 5-R. From Tank 5-7 the waste is transferred to Tank 15-1, via Tanks 5-8 and/or 5-9. The waste in Tank 15-1 is sampled, analyzed, chemically treated to meet storage specifications and, finally, sent to the West Area tank farms.

The headend waste, potentially radioactive, empties through a 6-in-diameter drain into the canyon 24-in-diameter sewer line.

#### 10.1.2 Types of Waste Generated

The T Plant liquid waste is composed primarily of potentially radioactive waste water from decontamination work performed in 2706-T, the canyon of T Plant, and the railroad tunnel. Small amounts of wastewater also come from steam condensate originating in the canyon.

Tank 15-1 is the central collection point for all T Plant liquid waste. It holds the waste until approximately 11,000 gal are accumulated, at which time its contents are transferred to the tank farms. The chemical contaminants in the tank waste are the cleansing agents introduced during decontamination work and the chemicals added to the waste in Tank 15-1 for pH and corrosion control. Table A.10-1 lists the volume of waste transferred from Tank 15-1 to the tank farms with a description of the waste constituents.

### 10.2 WASTE MINIMIZATION ACTIVITIES

The current waste minimization program at T Plant resulted in three specific activities that lead to minimization of liquid waste. One activity was the elimination of use of decontamination solutions containing potentially listed solvents, resulting in the elimination of the waste stream as F001, F002, or F003 (these designations indicate waste containing methylene chloride, 1,1,1-trichloroethane, or acetone).

Table 10-1. Waste Transfers to Tank Farms.

DATE	Volume	ACCUMULATED VOLUME	MnO4 (g/L)	Nitrite (PPM)	Hydroxide (g/L)	PH	Phosphate (g/L)	Sulfate (g/L)	Beta	Alpha	Pu (g/L)	Uranium (g/L)
01-Oct-87		0.00										
28-Jan-88	9371.35	9371.35	2.84 E-04	731.40	3.89 E-02	12.20	2.73 E-03	4.01 E-07	9.11 E+00	4.52 E-01	4.01 E-07	2.33 E-03
05-Feb-88	11872.30	21243.65	1.06 E-05	630.20	1.84 E-03	12.22	4.74 E-04	7.80 E-04	9.07 E+00	5.44 E-02	4.30 E-07	3.01 E-02
22-Feb-88	11956.00	33199.65	7.26 E-05	874.00	2.36 E-02	12.13	2.62 E-03	8.38 E-04	1.38 E+01	3.07 E-01	1.63 E-06	6.35 E-03
30-Mar-88	11543.20	44742.85	1.26 E-04	1224.00	3.75 E-02	12.21	1.54 E-03	5.97 E-04	3.29 E+01	4.04 E-01	2.33 E-06	1.04 E-02
13-Apr-88	11688.00	56430.85	1.17 E-05	727.00	6.48 E-02	12.62	9.22 E-04	3.93 E-04	1.28 E+01	5.16 E-01	9.59 E-06	8.13 E-03
19-Apr-88	11518.00	67948.85	1.71 E-05	653.20	6.08 E-02	12.49	8.39 E-04	4.09 E-04	1.62 E+01	3.38 E-01	3.19 E-07	1.39 E-02
26-Apr-88	10303.30	78252.15	1.73 E-04	745.20	6.58 E-02	12.71	6.25 E-04	8.36 E-04	1.46 E-02	5.92 E-05	4.01 E-07	1.46 E-02
06-May-88	11828.10	90080.25	1.29 E-04	745.20	4.10 E-02	12.41	9.73 E-04	1.20 E-03	5.65 E+01	4.69 E-02	1.42 E-05	
15-Jun-88	11695.00	101775.25	9.78 E-06	717.60	5.46 E-02	12.32	9.73 E-04	5.95 E-04	1.13 E+01	6.94 E-02	1.25 E-07	1.80 E-02
06-Jul-88	12579.60	114354.85	1.71 E-05	733.00	4.65 E-02	12.52		3.46 E-04	9.82 E+00	4.66 E-01	4.08 E-07	2.39 E-02
07-Jul-88	12229.00	126583.85	1.54 E-06	579.60	7.70 E-02	12.85		3.02 E-04	1.44 E+00			4.57 E-03
15-Jul-88	12315.00	138898.85	2.71 E-07	703.80	2.42 E-02	12.74	1.36 E-03	1.29 E-03	1.14 E+01	3.62 E-01		5.56 E-03
16-Jul-88	12316.00	151214.85	2.43 E-05	713.00	7.42 E-02	12.53	7.70 E-05	2.89 E-04	2.79 E+00	4.44 E-02	7.89 E-07	7.57 E-03
18-Jul-88	12654.00	163868.85	3.41 E-05	717.00	7.62 E-02	12.54	7.70 E-05	2.78 E-04	7.00 E+00	4.07 E-02	3.97 E-07	7.65 E-03
13-Aug-88	12059.00	175927.85	9.37 E-05	736.00	7.61 E-02	12.50	7.73 E-05	2.85 E-04	9.26 E+00	2.86 E-01	1.08 E-06	2.73 E-03
12-Sep-88	11905.00	187832.85	2.89 E-05	722.20	6.78 E-02	12.50	2.06 E-04	5.78 E-04	6.80 E-03	1.22 E-01	5.47 E-07	3.81 E-03
17-Nov-88	11513.25	199346.10	1.32 E-04	616.40	2.02 E-02	12.27	6.14 E-04	9.77 E-03	1.75 E+02	3.39 E-02	3.83 E-07	9.77 E-02
27-Dec-88	10320.90	209667.00		1175.00	5.65 E-02	12.74	2.76 E-04	2.75 E-04			9.93 E-08	3.70 E-02
30-Sep-89												

Another waste minimization activity was a procedure change to leave water in a well car after treating rather than pumping it out to send the car back empty for the next fuel shipment. This water is used for shielding at the fuel basins before each shipment. This activity resulted in the eliminating 5,000 gal of water per tank car which would have required storage and eventual evaporation.

Finally, a reduction in waste water resulted from a procedure change to directly measure the amount of shielding water added to the 20,000-gal-capacity waste tanker rather than a time and volume calculation that tended to add excess water. An estimated reduction in liquid waste water is 100 gal/tanker.

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## 11.0 HANFORD WASTE VITRIFICATION PLANT

### 11.1 DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED

#### 11.1.1 Description of Facility

The HWVP will be located in the 200 East Area of the Hanford Site. It will have the primary mission of converting high-activity radioactive and hazardous liquid wastes to a vitrified form (borosilicate glass) for disposal in a repository.

Currently, these wastes are stored in DSTs at the Hanford Site. The HWVP will also be designed to allow processing of SST wastes with some modifications. Tank wastes will be pretreated to separate the high-activity wastes from the low-activity wastes for processing by HWVP. The low-activity wastes will be treated in the Grout Treatment Facility (GTF). The HWVP includes the following five major activities: (1) feed receipt and preparation, (2) vitrification, (3) canister handling, (4) process offgas treatment, and (5) waste handling.

#### 11.1.2 Types of Waste Generated

The following general types of wastes are expected to be generated at the HWVP:

1. Process waste.
2. Decontamination waste.
3. Maintenance waste.
4. Miscellaneous waste.
5. Nonprocess waste.

### 11.2 WASTE MINIMIZATION ACTIVITIES

The HWVP waste minimization program will be tied to the overall waste minimization program for the Hanford Site. The HWVP waste minimization program will include all practices that reduce, avoid, or eliminate dangerous waste generation:

1. Minimize the volume of dangerous waste generated.
2. Generate nondangerous or less dangerous forms of waste, if practicable.
3. Segregate dangerous waste from nondangerous waste, if practicable.
4. Treat the dangerous waste forms to reduce toxicity.

A certification will be placed into the HWVP operating record on an annual basis, stating that a waste minimization program is in place. A Hanford Site-wide biennial report will be made to the U.S. Environmental Protection Agency (EPA), describing efforts to minimize waste and a certification that a waste minimization program is in place. Specific plans or procedures for waste minimization are described in the following sections.

#### 11.2.1 Process Waste

Process waste, consisting of secondary mixed waste, secondary nonradioactive waste and failed equipment, will be minimized by the following methods:

1. Secondary mixed waste will be segregated from nonradioactive dangerous waste and from nonradioactive, nondangerous waste to ensure that the volume of radioactive waste does not increase unnecessarily.
2. Secondary nonradioactive dangerous waste, resulting from spills or leaks, will be treated to reduce its volume and toxicity. Self-neutralization, to the extent practicable, of caustic compatible and acid waste to a pH of 2 to 14 will be used, reducing the need to add neutralizing agents which would increase the volume of waste to be disposed. Neutralized waste and related flush and washdown water will be transferred to a solar evaporator tank where the waste will be dewatered through natural evaporation, thus reducing waste volume. Further treatment of the waste will reduce the hazardous characteristics as a result of biological action and exposure to sunlight (i.e., components of the waste will decompose). Eventually, the collected residues will be disposed of at a permitted treatment, storage, or disposal facility.
3. Failed equipment will be repaired and returned to service if possible, reducing the quantity of equipment contaminated by hazardous waste. If repair is not feasible, the equipment will be packaged and disposed of in accordance with solid waste disposal procedures and requirements.

#### 11.2.2 Decontamination Waste

Decontamination waste, consisting of decontamination solid, nonrepairable equipment, and solid solution-absorbing waste, will be minimized through the following methods.

1. The repair or removal of failed equipment will require that radioactive material trapped or deposited within the equipment be removed by rinsing, washing, or wiping with decontamination solutions. The minimum quantity of decontamination solution required to achieve the required level of decontamination will be used. The decontamination effort will use the least toxic solution, such as water, first and then move to more toxic solutions as needed. In addition, efforts will be made to establish a purchasing procedure that will allow less toxic alternative products to be used on a trial basis to

evaluate their applicability to the particular decontamination process.

2. Small amounts of liquid decontamination solutions spilled or remaining in equipment will be removed using adsorbents such as paper, rags, and/or other solids. These adsorbents, containing mixed contaminants, will be analyzed, if required, and classified to determine the correct disposal process. Some of this waste may be reduced in volume by use of a compactor. Waste that is shown by analysis to be nondangerous will be segregated from the dangerous waste and disposed of at a landfill.

### 11.2.3 Maintenance Waste

Maintenance waste, consisting of substances such as lubricating oil, antifreeze, oil absorbent, and solvent, will be minimized in the following manner:

1. Oil changes will be scheduled at maximum allowable intervals, reducing the quantities of used oil generated.
2. Antifreeze will be analyzed to determine if it performs as required. If a specific property is found to be outside the operating specifications, the property will be altered, if possible, to bring it within specifications, by using an additive.
3. Review of maintenance materials, such as solvents, will be made to determine if a less toxic product can be substituted, reducing the volumes of dangerous materials employed and the inventories of maintenance materials required.
4. Waste will be segregated where possible. Antifreeze and oil will not be mixed, making refinement of waste oil less costly and more attractive to recyclers.
5. A program to recycle contaminated oil will be developed, reducing the disposal volumes of waste.

### 11.2.4 Miscellaneous Waste

Miscellaneous waste usually results from oils contaminated with wastes or from off-normal events such as spills or equipment failures. Waste minimization will assess the methods best suited for the cleanup of spills.

### 11.2.5 Nonprocess Waste

Nonprocess waste, such as office refuse, will be segregated from the more dangerous waste and disposed of in an onsite landfill, reducing the volumes of waste that must be treated as dangerous waste.

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## 12.0 GROUT TREATMENT FACILITY

### 12.1 DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED

#### 12.1.1 Description of Facility

The GTF, located in the 200 East Area of the Hanford Site, has the primary mission of permanently disposing of LLW. These LLWs will be blended with cementitious materials for immobilization and solidification in below-ground vaults. The GTF includes the Dry Materials Facility (DMF), the Grout Processing Facility (GPF), and the Grouted Waste Disposal Facility (GDF).

The DMF has the primary purpose of receiving, storing, and blending the dry cementitious grout materials. Materials used in this facility include portland cement, fly ash, and blast furnace slag. No radioactive materials are handled at the DMF.

The GPF has the primary purpose of receiving radioactive liquid LLW from the 241-AP Tank Farm feed tank, mixing it with the dry-blend materials from the DMF, and transferring the resulting grout mixture to a disposal vault.

The GDF is where the Grout Disposal Vaults are located. The grout slurry mixture is pumped into the vault and cures into a hardened grout product. Liquid waste generated by the grout process or excess water and leachate liquid from the vault during the setting and curing process is returned to the tank farms for processing. Flush liquids result in additional liquid waste.

#### 12.1.2 Type of Waste Generated

The tank waste the GTF has generated is a low-activity radioactive and hazardous liquid waste (approximately 52,000 gal during 2 yr).

### 12.2 WASTE MINIMIZATION ACTIVITIES

The waste minimization plan has the primary purpose to reduce the volume, weight, or toxicity of all regulated waste generated at GTF to the extent practical. Areas addressed in the plan include organizational responsibilities, employee training, employee participation and incentive programs, and incorporation of waste minimization as part of the design process for new projects or designs.

#### 12.2.1 Employee Training

As part of general training for new employees, waste minimization training is included. General waste minimization training is provided to all employees of the GTF via waste minimization team awareness presentations and for hazardous waste shippers as part of the "Hazardous Waste Shipment Certification" class. Specific training and application of waste minimization

techniques will be provided on an individual or group basis, as appropriate, by the respective manager or supervisor. The manager or supervisor is responsible for establishing employee responsibilities, assignments, and goals. Each group will keep a record of waste minimization training.

#### 12.2.2 Employee Participation and Incentive Program

An employee participation and incentive program is part of the waste minimization plan at the GTF. Promotion and application of employee incentives appear to be a good way to minimize waste generation and maximize the use of good operating procedures. The incentive program has several components:

- Encourage employees to submit suggestions as Price proposals or Great Ideas.
- Encourage employees to submit suggestions to the Westinghouse Hanford waste minimization specific incentive program (currently being developed).
- Encourage employees to submit "on-the-job" type waste minimization ideas directly to the GTF Waste Minimization Team with certificates and other "thanks" for this program.

#### 12.2.3 New Projects and Designs

New projects and designs will be required to include waste minimization as an integral part of the design process. To accomplish this, the GTF waste minimization representative will review any proposed new construction and major grout process changes to ensure that waste minimization has been considered. New construction presently under consideration include (1) four Grout Disposal Vaults, (2) modification to Tank 241-AP-104 for use as a second feed tank, and (3) a Grout Failed Equipment Handling Facility to stage contaminated failed equipment.

APPENDIX B

EVALUATION AND SELECTION OF BOROSILICATE  
GLASS AS THE WASTE FORM FOR HANFORD  
SITE HIGH-LEVEL RADIOACTIVE WASTE

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**A White Paper**

**Evaluation and Selection of  
Borosilicate Glass as the Waste  
Form for Hanford High-Level  
Radioactive Waste**

May 1990

Prepared by the Project Technical Support Office

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A White Paper

EVALUATION AND SELECTION OF BOROSILICATE GLASS  
AS THE WASTE FORM FOR HANFORD HIGH-LEVEL RADIOACTIVE WASTE

May 1990

Prepared by the Project Technical Support Office

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EVALUATION AND SELECTION OF BOROSILICATE GLASS  
AS THE WASTE FORM FOR HANFORD HIGH-LEVEL RADIOACTIVE WASTE

EXECUTIVE SUMMARY

Borosilicate glass is the waste form of choice for solidification of high-level radioactive waste worldwide. Thirteen production plants are in operation or under construction for the conversion of liquid high-level waste to borosilicate glass. Eleven of the plants are in foreign countries; two are in the U.S., at the Savannah River Site and at West Valley. The Hanford Waste Vitrification Plant (HWVP) will be the third U.S. plant for the conversion of high-level nuclear waste to borosilicate glass.

The evaluation of alternative waste forms that reached its peak in the late 1970's has largely abated because of the advanced state of development and utilization of borosilicate glass. Synroc, a hot-pressed crystalline waste form, is still being developed in Australia. A large nonradioactive Synroc pilot plant is in operation, but the Synroc process has not yet been tested radioactively, except for a few laboratory-scale experiments. Synroc, or one of the many other alternatives to borosilicate glass that have been proposed, could emerge as a potentially feasible second generation waste form for use some time in the future, but numerous questions remain to be answered. In the meantime, many studies have shown the suitability of borosilicate glass.

The selection of borosilicate glass as the waste form at the three U.S. sites was made only after a thorough technical review of the alternatives and after public comment through the National Environmental Policy Act (NEPA) process. There has been an important evolution in disposal regulations and applicable tests for waste forms since 1982. Recently, questions have arisen concerning whether the original waste form selection process is still valid for HWVP, given the Resource Conservation and Recovery Act (RCRA) requirements and other considerations. The purpose of this white paper is to summarize the evolution in waste disposal regulations and nuclear waste form development that have occurred since the early 1980's, and to assess the impact of this evolution on the appropriateness of the selection of borosilicate glass as the waste form for the HWVP. A review of borosilicate glass data as they apply to the new waste form tests and evolving regulations shows that borosilicate glass remains a fully satisfactory waste form for high-level waste immobilization in the HWVP. In addition, the Environmental Protection Agency (EPA) has recently promulgated vitrification as the treatment standard, i.e., best demonstrated available technology (BDAT), for the high-level fraction of the mixed waste generated during the reprocessing of nuclear fuel (Federal Register, June 1, 1990).

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CONTENTS

INTRODUCTION . . . . .	1
CHRONOLOGY OF EVENTS LEADING TO THE SELECTION OF BOROSILICATE GLASS AS THE WASTE FORM AT DWPF, WVDP, AND HWVP . . . . .	1
Early Waste Form Investigations . . . . .	2
National High-Level Waste Technology Program . . . . .	2
Selection of Borosilicate Glass at DWPF, WVDP, and HWVP . . . . .	5
RECENT WASTE FORM QUALIFICATION ACTIVITIES . . . . .	11
Waste Acceptance Preliminary Specifications . . . . .	11
Leach Tests To Demonstrate Compliance with DOE Specifications . . . . .	12
Leach Tests To Demonstrate Compliance with EPA Specifications . . . . .	13
Leach Tests To Improve Understanding of Borosilicate Glass Behavior in a Nuclear Waste Repository . . . . .	13
INTERNATIONAL BOROSILICATE GLASS TECHNOLOGY . . . . .	14
France . . . . .	15
Other European Countries . . . . .	15
Non-European Countries . . . . .	16
ALTERNATIVE WASTE FORM ACTIVITIES . . . . .	17
Glass-Ceramic . . . . .	17
Phosphate Glass . . . . .	18
Synroc . . . . .	18
SUMMARY AND CONCLUSIONS . . . . .	20
BIBLIOGRAPHY . . . . .	23
<u>The Evaluation and Selection of Candidate High-Level Waste Forms, DOE/TIC-11611, March 1982 . . . . .</u>	<u>25</u>
<u>The Waste Acceptance Preliminary Specifications . . . . .</u>	<u>26</u>

CONTENTS (Contd)

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<u>A Study of the Isolation System for Geologic Disposal of Radioactive Wastes</u> , National Academy Press, April 1983 . . . . .	28
Lutze, W., and R. C. Ewing, <u>Radioactive Waste Forms for the Future</u> , North-Holland, 1988 . . . . .	29

INTRODUCTION

Following thorough evaluations of the alternatives in the late 1970's and early 1980's, decisions were made to solidify high-level radioactive wastes at Savannah River, West Valley and Hanford in borosilicate glass. The high-level wastes at the three sites are similar, all coming from nuclear fuel reprocessing, and all being neutralized except one tank at West Valley; however there will be variations in the waste streams being solidified. The ability of borosilicate glass to accommodate variations in waste composition, plus the advanced state of development of vitrification technology, made it the waste form of choice at each site.<sup>1</sup> Facilities for the production of borosilicate glass in the U.S. are constructed, under construction, or in Title II design, at Savannah River, West Valley and Hanford, respectively. These facilities are the Defense Waste Processing Facility (DWPF), the West Valley Demonstration Project (WVDP), and the Hanford Waste Vitrification Plant (HWVP).

Recently, questions have arisen concerning whether the original waste form selection process is still valid for HWVP, given the Resource Conservation and Recovery Act (RCRA) requirements and other considerations. The purpose of this white paper is to summarize the evolution in waste disposal regulations and nuclear waste form development that have occurred since the early 1980's, and to assess the impact of this evolution on the appropriateness of the selection of borosilicate glass as the waste form for the HWVP.

The Environmental Protection Agency (EPA) has recently issued its final rulemaking for Land Disposal Restrictions in the Federal Register dated June 1, 1990. In this rulemaking, EPA has promulgated vitrification as the treatment standard, i.e., best demonstrated available technology (BDAT), for the high-level fraction of the mixed waste generated during the reprocessing of nuclear fuel. The hazardous components in the mixed waste are regulated by EPA under the RCRA, and the radioactive components are regulated under the Atomic Energy Act. In establishing this ruling, EPA concluded that vitrification will provide effective immobilization of the inorganic RCRA hazardous constituents in high-level mixed waste generated during the reprocessing of fuel rods.

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<sup>1</sup> Because waste variations produce variations in the final glass composition, the term borosilicate glass as used in this white paper does not denote a single composition, but rather a family of glasses designed for the incorporation of nuclear waste, whose principal matrix components are silicon and boron oxides.

CHRONOLOGY OF EVENTS LEADING TO THE SELECTION OF BOROSILICATE GLASS AS THE  
WASTE FORM AT DWPF, WVDP, AND HWVP

Research and development of glass as a waste form<sup>2</sup> for the immobilization of liquid radioactive wastes began in the mid-1950s.<sup>2</sup> Thus the decisions, over 25 years later, to convert the high-level wastes at Savannah River, West Valley, and Hanford to glass were based on a large store of information and a long legacy of investigation and research.

Early Waste Form Investigations

Man's experience in the manufacture and utilization of glass for many different purposes dates from pre-Roman times. Early investigators recognized silicate glasses as a promising medium for the immobilization of liquid radioactive wastes because they were chemically inert, thermally stable, capable of incorporating many different elements and technologically straight forward to process. Furthermore, there was significant evidence of the long-term durability of these glasses based on the existence of natural analogs of these materials in the earth's crust.

The investigators in the 1950s envisioned a batch process in which the radioactive waste and glass making components would be melted in ceramic crucibles. The glass compositions were high-melting ( $>1300^{\circ}\text{C}$ ). The investigations were centered at the Massachusetts Institute of Technology in the U.S. and Chalk River in Canada. It was soon recognized that the use of lower-melting glasses would make remotely-operated processing more efficient by decreasing off-gas treatment requirements and increasing melter lifetime. Borosilicate glass formulations produced the desired lower melting ( $<1150^{\circ}\text{C}$ ) glasses.

Radioactive pilot plant demonstrations of borosilicate glass waste immobilization processes were conducted during the 1960s in England, France, and the U.S. This radioactive pilot plant development work led to the world's first facility for the production of borosilicate waste glass, which has operated continuously at Marcoule, France, since July 1978.

Two major radioactive test programs to produce borosilicate waste glass were completed in the U.S. for the Department of Energy (DOE) by the Pacific Northwest Laboratory (PNL) between 1969 and 1979. These programs demonstrated the in-can melting process for producing borosilicate glass.

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<sup>2</sup> The U.S. DOE and its predecessor organizations have a long history of concern about the handling of radioactive wastes. An early record of this is a seminar, co-sponsored by the Atomic Energy Commission (AEC) and the Public Health Service, held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955. The 609-page proceedings of this meeting (TID-7517) contains several descriptions of waste form research, including the firing of clay-flux mixtures containing radioactive wastes to produce a glass-like waste form.

National High-Level Waste Technology Program

Although the French had begun production of borosilicate waste glass, and processes for its production had been developed by DOE, it was decided to thoroughly review the alternatives before making a final decision on the waste form to be used for the immobilization of high-level waste in the U.S.

This was accomplished through DOE's National High-Level Waste Technology Program conducted between 1979 and 1981,<sup>3</sup> and by assessments performed separately at three DOE sites and West Valley.

The National High-Level Waste Technology Program sponsored research and development on proposed waste forms at 14 laboratories, at three universities, three industrial laboratories, and several DOE sites. Seventeen different candidate waste forms were considered. Development activities on ten of the waste forms were terminated as the program proceeded when preliminary reviews raised technical concerns about the viability of these forms as candidates for geologic disposal of wastes.

The DOE through its High-Level Waste Lead Office located at Savannah River Laboratory (SRL) undertook an evaluation of the final seven candidate waste forms. These evaluations are described below:

- An evaluation of the waste form product properties considered to be important to waste immobilization and geologic disposal was performed at SRL.
- A processability analysis was conducted at the Engineering Department of E. I. duPont de Nemours and Company in Wilmington, Delaware.
- Waste-specific evaluations were conducted by the DOE sites at Savannah River, the Idaho National Engineering Laboratory (INEL) and Hanford.

And finally, to obtain an independent review, an Alternative Waste Form Peer Review Panel chaired by Dr. L. L. Hench, of the University of Florida, was organized by DOE to evaluate the alternative waste forms. The peer review panel membership was from non-DOE research organizations, representing a cross section of university, industry and government laboratories from disciplines of materials science, ceramics, glass, metallurgy, and geology. Four meetings were held at which participants in the National High-Level Waste Technology Program presented data on proposed waste forms to the panel. The panel

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<sup>3</sup> The DOE, and its predecessor organizations had previously conducted an extensive investigation of alternative waste forms for commercial high-level waste. These investigations culminated in the Waste Solidification Engineering Prototypes (WSEP) program at PNL. The WSEP program was a fully radioactive pilot plant program in which four candidate immobilization processes (pot calcine, phosphate ceramic, phosphate glass, and in-can melting of borosilicate glass) were demonstrated in the years 1966 through 1970.

considered leach resistance, waste loading, mechanical strength, radiation stability, and thermal stability, and documented their conclusions in three reports. In the third and final report they ranked the waste forms, finding borosilicate glass and Synroc to rank one and two, respectively.

The alternative waste forms were ranked using weighted input from all of the evaluations described above and the conclusions from the peer review panel. This ranking was reported in DOE/TIC-11611 (see bibliography). The scope of this ranking assured that processability was given due consideration, along with waste form characteristics. The ranking of the final seven candidate waste forms was:

1. Borosilicate glass
2. Synroc
3. Tailored ceramic
4. High-silica glass
5. FUETAP concrete
6. Coated particles
7. Glass marbles in a lead matrix

Detailed descriptions of alternative waste forms are beyond the scope of this white paper, but a brief discussion of the seven top-ranked candidates is warranted to give perspective on the breadth of the kinds of waste form materials that have been evaluated.<sup>4</sup>

**Borosilicate Glass** - A predominantly noncrystalline, relatively homogeneous waste form in which individual atoms of waste are randomly distributed and bonded to a silicon-based network. Homogeneity is achieved during processing, when the glass is held in the molten state at 1050-1150°C before casting in stainless steel canisters.

**Synroc** - A predominantly crystalline waste form first proposed by Dr. A. E. Ringwood of the Australian National University in 1978. The strategy is to use naturally occurring minerals, known to survive for long times in nature, as prototypes for synthetic minerals in which the various waste constituents are incorporated. An assemblage of three or more synthetic minerals is usually required to incorporate all of the waste constituents. Dr. Ringwood first proposed crystallizing the minerals from a >1350°C melt, but this was replaced by hot isostatic pressing.

**Tailored Ceramics** - A predominantly crystalline waste form mainly investigated at Rockwell Science Center, but with contributions from other laboratories such as Pennsylvania State University. This waste form is somewhat similar to Synroc; however, selection of the crystalline phases to be hosts for the waste components is based more on materials science than geological considerations. The result is that, whereas Synroc emphasizes

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<sup>4</sup> There is a considerable literature on radioactive waste forms. A good introduction to the literature is the series, "Scientific Basis for Nuclear Waste Management," published annually by the Materials Research Society, Pittsburgh, Pennsylvania, since 1979.



titanium-containing phases, the tailored ceramics studied have consisted of four to six nontitanium phases. The peer review panel concluded that tailored ceramics showed promise as a waste form but questioned the complexity of the formulations.

**Porous Glass Matrix (high-silica glass)** - A somewhat heterogeneous waste form, produced by sintering an intimate mixture of calcined waste and porous glass powder (10 to 300- $\mu$ m diameter grains with 0.01- $\mu$ m pores) at 1200°C. A high silica and alumina content can be achieved in the final glass, producing very good leach resistance. The waste is not totally dissolved in the glass; rather the waste particles tend to remain as discrete particles dispersed in the high-silica glass matrix.

**Concrete (FUETAP: formed under elevated temperature and pressure)** - A heterogeneous crystalline waste form. Waste, water, cement, and other solid powder additives are mixed and cured to form a monolith. When the curing is accomplished under elevated temperatures (100 to 250°C) and pressures (1000 psi) for 24 hours, the product is termed FUETAP. However, the processing proposed for high-level wastes is somewhat different: the cement-waste mixture is poured into the canister and cured under mild autoclave conditions (100°C, 1 atm steam). Subsequently, the concrete is dewatered under vacuum at 250°C for 24 hours.

**Coated Sol-Gel Particles** - A heterogeneous waste form consisting of small ( $\leq 1$  mm), predominantly crystalline, waste-containing particles coated with three layers (low-density SiC, high-density SiC, and high-density pyrolytic C) and embedded in an inorganic binder (probably an aluminate or silicate-aluminate compound). The fabrication of this waste form is largely based on technology developed for fabrication of high-temperature gas reactor fuel. The sol-gel process is used to fabricate spherical particles of a waste-containing formulation. Cesium cannot be retained in the sol-gel particles and is separately adsorbed on zeolite particles. The sol-gel and zeolite particles are then coated with SiC and pyrolytic C in a fluidized bed reactor.

**Glass Marbles in a Lead Matrix** - A heterogeneous waste form consisting of borosilicate glass marbles (~1.3 cm diameter) embedded in a continuous matrix of lead-tin alloy. Because the glass marbles can make up no more than 60% of the volume of the waste form, the waste loading is decreased significantly in comparison with a canister filled with borosilicate glass. The high heat conductivity of the lead matrix could be advantageous for commercial nuclear wastes, but is of little value for the low-heat producing defense wastes.

#### Selection of Borosilicate Glass at DWPF, WVDP, and HWVP

The evaluations conducted by the National High-Level Waste Technology Program documented the fact that borosilicate glass was the most well-developed and viable waste form for both defense and commercial wastes in 1982. Commercial wastes were of interest because the federal government shortly prior to that time had committed to solidify the liquid high-level wastes remaining at the closed-down commercial nuclear fuel reprocessing plant at West Valley, New York.

The waste compositions at West Valley, Savannah River and Hanford sites vary among the three sites as well as within the sites. West Valley will have the smallest variation in the waste composition. The relatively small amount of waste at West Valley can be well characterized and processed as one batch. The much larger quantities of waste at Savannah River and Hanford will have to be processed in many batches coming from many different storage tanks. The composition of the waste batches will differ most markedly at Hanford where several different reprocessing methods have been employed, some of the wastes have already been treated to remove strontium and cesium, and some of the wastes resulted from other chemical operations. Examples of the ranges in waste compositions to be vitrified are shown in Table 1. The evaluations of candidate waste forms performed by the three sites and the National High-Level Waste Technology Program all found that the capability of borosilicate glass to readily accommodate fluctuations in waste composition was a significant advantage over competing candidate waste forms. Another major advantage was that remote-operation of the processing equipment to produce borosilicate glass was successfully demonstrated technology.

The U.S., as well as other countries having liquid high-level nuclear waste, were moving forward with the immobilization of these wastes prior to identifying a final disposal location. Final disposal locations had not (and have not) been identified, thus waste form behavior under repository-specific disposal conditions could not be studied. However, the dissolution rates of the candidate waste forms in deionized water (and some simulated potential repository waters) were compared as part of the selection process. It was found that the dissolution rates (commonly called leach rates) of borosilicate glasses were higher than some candidate waste forms, and lower than others.

TABLE 1. Typical High-Level Waste Vitrification  
Feed Compositions

Element	Concentration Ranges (wt%)*		
	DWPF(1)	WVDP(1)	HWVP(2)
Iron	3-37	1-36	21-44
Aluminum	2-40	---	9-11
Manganese	3-7	1	3
Uranium	2-10	0-5	3-6
Sodium	2-6	1	2-8
Calcium	1-4	---	0.4-2
Nickel	0.5-6	1	0-2
Thorium	---	0-39	---

\* Will not sum to 100%.

- (1) Analysis of the Terminal Waste Form Selection for the West Valley Demonstration Project, WVDP-100, Table 2, 1984.
- (2) EIS - Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, DOE/EIS-0113, vol. 2 of 5, pp. C.13-14, December 1987.

In other words, leachability was an important, but not exclusive, criterion for the selection of borosilicate glass.

Chronologies for the selection of borosilicate glass at the DWPF, WVDP, and HWVP are shown in Tables 2, 3, and 4, respectively. Similar procedures were followed at the three sites, namely, a technical evaluation of the candidate waste forms, followed by use of the National Environmental Policy Act (NEPA) process to obtain public comment on the waste immobilization process selected. In June, 1983, the U.S. DOE issued the "Defense Waste Management Plan," DOE/DP-0015, which states, "The objective is to end interim storage and to achieve permanent disposal by immobilizing and preparing high-level waste for shipment to a geologic repository," and "This approach permits the experience gained at the first site [i.e., DWPF] to be applied to the other sites."

Facilities for the production of borosilicate glass in the U.S. are at various stages of completion. Groundbreaking for the DWPF was initiated in 1983 with construction to be completed in 1990. At West Valley, fabrication of process equipment, including the melter for the Component Test Stand (CTS), was initiated in 1984. After cold-testing, the CTS will become part of the future process system. Extensive testing of the melter in the CTS was completed in 1989 and construction of cell walls and process building around the CTS was initiated in 1990. At Hanford, the Title II design of HWVP was initiated in 1990.

**TABLE 2.** Chronology for Selection of Borosilicate Glass  
at the Defense Waste Processing Facility (DWPF)

1977	<p>Conceptual design of the DWPF starts.</p> <p><i>Borosilicate glass was used as the design reference waste form based on extensive U.S. and international studies.</i></p>
1982	<p>"Final Environmental Impact Statement, Defense Waste Processing Facility," Savannah River Plant, Aiken, SC," DOE/EIS-0082, February, 1982.</p> <p><i>The environmental consequences of selecting borosilicate glass are small.</i></p> <p>"The Evaluation and Selection of Candidate High-Level Waste Forms," DOE/TIC-11611, March, 1982.</p> <p><i>Candidate waste forms were ranked using inputs from four kinds of evaluations, applicability to defense waste compositions, product performance, processability, and an independent peer review. Borosilicate glass led the combined rankings, with Synroc second.</i></p> <p>"Defense Waste Processing Facility, Savannah River Plant, Aiken, S.C.; Record of Decision," <u>Federal Register</u>, Vol.47, No. 105, pp. 23801-23803, June 1, 1982.</p> <p><i>Proposed action is to select borosilicate glass as the DWPF waste form.</i></p> <p>"Environmental Assessment, Waste Form Selection for SRP High-Level Waste," DOE/EA-0179, July, 1982.</p> <p><i>The potential environmental consequences of the selection of borosilicate glass as the reference waste form for the DWPF, vs. an alternative waste form, Synroc-D (a Synroc specially formulated for Savannah River wastes), are described.</i></p>
1983	<p>"Compliance With the National Environmental Policy Act Proposed Finding of No Significant Impact, Selection of Borosilicate Glass as the Defense Waste Processing Facility Waste Form for High-Level Radioactive Wastes Savannah River Plant, Aiken, South Carolina," <u>Federal Register</u>, Vol. 47, No. 146, 32778-32783, July 29, 1982.</p> <p><i>Borosilicate glass is the waste form.</i></p>

TABLE 3. Chronology for Selection of Borosilicate Glass  
at the West Valley Demonstration Plant (WVDP)

- 1978 "Western New York Nuclear Service Center Study." TID-28905.  
U.S. Department of Energy, Washington, D.C.
- Study performed under DOE direction by several subcontractors under the lead of Argonne National Laboratory to evaluate options for decommissioning the West Valley site. Vitrification was proposed as one method for immobilizing the liquid high-level waste.*
- 1980 Public Law 96.368 "The West Valley Demonstration Act."
- Authorizes DOE to carry out a demonstration of the solidification and preparation for disposal of liquid high-level radioactive waste.*
- 1981 "Draft Environmental Impact Statement, Long-Term Management of Liquid High-Level Radioactive Waste Stored at the Western New York Nuclear Service Center, West Valley," DOE/EIS-00810, July 1981.
- Borosilicate glass was used as the reference solidified form for the EIS study.*
- 1982 "Final Environmental Impact Statement, Long-Term Management of Liquid High-Level Radioactive Waste Stored at the Western New York Nuclear Service Center, West Valley," DOE/EIS-00810, July 1982.
- Borosilicate glass is the reference waste form.*
- 1984 "Analysis of the Terminal Waste Form Selection for the West Valley Demonstration Project," WVDP-100.
- Reasons that borosilicate glass is preferred over 16 other waste forms are documented.*

TABLE 4. Chronology for Selection of Borosilicate Glass at the Hanford Waste Vitrification Plant (HWVP)

- 1977-78 "Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Hanford Reservation," ERDA-77-44, 1977, and "Radioactive Wastes at the Hanford Reservation - A Technical Review," an evaluation by the National Academy of Sciences' Committee on Radioactive Waste Management, 1978.

*These documents concluded that, although interim operations were being carried out in a safe and responsible manner, it was time to move ahead with the final disposal of Hanford wastes.*

- 1980 Schulz, W. W., et al. "Preliminary Evaluation of Alternative Forms for Immobilization of Hanford High-Level Defense Wastes," RHO-ST-32.

*A preliminary evaluation of alternative forms for immobilization of Hanford High-Level Defense Wastes. Nineteen alternative forms were considered and evaluated statistically against a set of predetermined criteria. The study concluded that borosilicate glass, concrete/bitumin, and ceramic waste forms were the top candidates for immobilizing the Hanford High-Level Waste.*

- 1987 "Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes," DOE/EIS-0113.

*The preferred alternative is to convert the sludge from existing and future double shell tank wastes to borosilicate glass; decision on the single shell tank wastes is deferred.*

- 1988 "Disposal of Hanford Defense High-Level, Transuranic, and Tank Wastes," Hanford Site, Richland, Washington; Record of Decision (ROD), Federal Register, Vol. 53, No. 72, pp. 12449-12453, April 14, 1988.

*Borosilicate glass is the waste form.*

RECENT WASTE FORM QUALIFICATION ACTIVITIES

There is a long background of governmental policy concerning nuclear waste prior to 1982 that is outside the scope of this white paper. Important examples in this policy background are the requirement in 1970 that commercial high-level waste be solidified within five years after generation and transferred to the Atomic Energy Commission (AEC) within ten years after generation (10 CFR 50, Appendix F), and the requirement in 1973 that defense "High-level liquid wastes shall be converted to suitable physical and chemical forms and confined in a manner which shall provide high assurance of isolation from man's environment with minimal maintenance and surveillance by man under conditions of credible geologic, seismic, and other naturally occurring events." (AEC Manual Chapter 0511).

Regulations applying to the disposal of commercial high-level nuclear wastes in geologic repositories were first promulgated in 1983 and 1985 by the U.S. Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA), respectively. A Presidential decision in 1985 permitted the commingling of defense wastes in a commercial high-level waste repository, and thus made defense wastes subject to the NRC and EPA regulations. The NRC and EPA regulations have little specificity regarding the waste form. Their concern is with radioactive release from the "engineered barrier system" (NRC) and to the "accessible environment" (EPA).

The NRC regulations endorse a multiple barrier waste package concept to allow flexibility in design of the waste disposal system. The regulations do not require that primary reliance be placed on the inertness of the waste form. Reliance can be placed on other barriers within the engineered barrier system, or on a combination of the waste form and other barriers. The latter strategy has been adopted for the candidate geologic repository above the water table in Yucca Mountain, Nevada. Metal containers, installed around the canisters of borosilicate glass before they are emplaced in the repository, provide a primary barrier for 300-1,000 years. Then the low solubility of the glass becomes a barrier as the metal containers and canisters begin to fail in a random fashion. Some cracking of the glass "logs" occurs during cooling and handling of the canisters, resulting in a surface area that may be substantially greater than if the glass logs were truly monolithic. Site characterization studies and engineered barrier system design will determine the extent to which the glass itself must be relied upon as a barrier.

Waste Acceptance Preliminary Specifications

Within DOE, basic responsibility for meeting the NRC and EPA regulations for geologic disposal lies with the Office of Civilian Radioactive Waste Management (OCRWM) and not with the waste producers. In order to carry out its responsibilities, OCRWM has drafted Waste Acceptance Preliminary Specifications (WAPS, see Bibliography). The intent is that canisters of borosilicate glass produced at the DWPF and WVDP (and ultimately HWVP) must satisfy these specifications before they can be received by the repository.

The four major WAPS specifications that apply to the waste form are:

- 1.1 Chemical Specification - Requires documentation of the chemical composition of the waste forms sent to the repository.
- 1.2 Radionuclide Inventory Specification - Requires documentation of the radionuclide content of the waste forms sent to the repository.
- 1.3 Specification for Radionuclide Release Properties - Requires that the waste form sent to the repository not exceed an established leach rate limit. Documentation is to be provided that 95% of the waste form meets the criterion at the 95% confidence level.
- 1.4 Specification for Chemical and Phase Stability - Requires that the glass transition temperature of the waste form sent to the repository be documented, and that documentation be provided that that temperature has not been exceeded during handling and transportation of the waste glass canisters prior to receipt at the repository.

Testing and analyses done at Savannah River, West Valley, and Hanford show that a borosilicate glass waste form can comply with the WAPS specifications. Only one of the four waste form specifications, WAPS Specification 1.3, provides a quantitative measurable limit. As such, it warrants the more detailed discussion below.

#### Leach Tests to Demonstrate Compliance with DOE Specifications

The purpose of WAPS Specification 1.3 is to ensure that the radionuclide release properties of the glass, i.e., the "quality" of the glass, have been controlled during production. Leach resistance has been established as a measure of quality by OCRWM. Glasses that demonstrate satisfactory leach resistance in distilled water will be tested at a later date under repository-specific conditions to establish release characteristics that can be used in repository performance assessments. Leach test data may also be needed in transportation and repository preclosure accident analyses.

The leach test specified in WAPS Specification 1.3 is the MCC-1 Static Leach Test conducted in deionized water at 90°C. The test duration is to be 28 days. The acceptance criterion is that the normalized elemental leach rate for the matrix elements sodium, silicon, and boron and for the radionuclides cesium-137 and uranium-238 shall be less than 1 g/m<sup>2</sup>/day averaged over the 28-day test duration. Borosilicate glass can meet this criterion (PNL-6723, see Bibliography).

Specification 1.3 permits the producer to propose an alternative approach to compliance. The DWPF has proposed testing the glass using an alternative test method called the product consistency test (PCT). The main difference between the MCC-1 leach test and the PCT is the test specimen. The MCC-1 leach test uses a monolithic specimen that must be sawed or otherwise shaped to have known dimensions. The PCT uses a 100 to 200 mesh crushed glass specimen. Both tests use deionized water at 90°C as the leachant, but the duration of the PCT is 7 days, rather than the 28-day length of the MCC-1 leach test. In



comparison with the MCC-1 test, the PCT is easier to conduct remotely in a hot cell, and most importantly, appears to offer substantially greater precision in terms of measured leach rates. This is a particularly important advantage with respect to the number of samples required to demonstrate compliance with the "95/95" specification.

#### Leach Tests to Demonstrate Compliance with EPA Specifications

The EPA has defined two extraction procedures, which may be thought of as accelerated leach tests, to distinguish whether or not a waste is chemically hazardous. Based on preliminary results obtained with these procedures, defense high-level waste immobilized in borosilicate glass is not a Resource Conservation and Recovery Act (RCRA) hazardous waste.

The two EPA leach tests, the Extraction Procedure Toxicity Test (EP Tox) and the Toxicity Characteristic Leaching Procedure (TCLP), are quite similar for inorganic waste constituents, and only inorganic waste constituents are of concern here. (The EPA is phasing out EP Tox; TCLP will be used exclusively after March 29, 1991.) Borosilicate glass will not retain organic waste constituents; they are decomposed or volatilized during melting. TCLP tests have been conducted on seven simulated DWPF glasses. EP Tox tests have been conducted on one simulated DWPF glass and three simulated West Valley glasses. The DWPF tests were performed by Environmental & Chemical Sciences, Inc., Aiken, South Carolina. The West Valley tests were performed at PNL, Richland, Washington and RECRA Environmental, Inc., Amherst, New York. The yet-to-be-published results show that the concentrations of the eight hazardous elements (arsenic, barium, cadmium, chromium, lead, mercury, selenium, and silver) in the leachates were below RCRA hazardous concentrations and usually below detection limits. In addition, the fact that geologic disposal is mandated for high-level radioactive wastes means that they will be handled in a more restrictive manner than any chemically hazardous wastes.

#### Leach Tests to Improve Understanding of Borosilicate Glass Behavior in a Nuclear Waste Repository

It is extremely difficult, if not impossible, to obtain leach test data in the laboratory that exactly represent repository behavior. The problem of relating leach test results to repository behavior is summarized in the following statement,

"... the dissolution and leaching rates of minerals and glasses under field conditions ... are consistently lower than rates measured in the laboratory, sometimes by several orders of magnitude. These differences are presumably due to processes not readily studied by short-term experiments in the laboratory but that are rate controlling in the long term (e.g., volume diffusion, surface film diffusion, or solubility-limited transport)." (page 300 in A Study of the Isolation System for Geologic Disposal of Radioactive Wastes, see Bibliography).

Some tests with borosilicate glass in simulated Yucca Mountain Repository conditions are being conducted by J. K. Bates at Argonne National Laboratory.

The purpose of these tests is not to define the leach rates that will pertain in the repository, but to improve understanding of the important reaction mechanisms. For the proposed Yucca Mountain Site, the expected repository conditions include contact with humid air and potential contact with small amounts of trickle-through water. Less likely conditions include contact with large volumes of standing water. With humid air contact, glass will very slowly transform into secondary mineral phases more stable than the glass. The stable mineral phases that form depend on the glass composition, but generally include zeolites, clays, and calcium silicates, which are capable of incorporating radionuclides into their structure. If humid air contact is followed at some time later by liquid water contact, release of radionuclides will be a function of both the secondary mineral phases and the aged glass matrix.

However, performance assessment of repository behavior does not require precise waste form release data, i.e., leach rate or particulate dispersion data, to show compliance with NRC and EPA requirements. Preliminary assessments show that repository performance is relatively insensitive to waste form behavior. This is because when the whole repository and its surrounding environment are taken into account, hydrological and geochemical processes external to the waste form dominate the analysis. A description of the methodology used in performance assessment is given in A Study of the Isolation System for Geologic Disposal of Radioactive Wastes (see Bibliography).

### INTERNATIONAL BOROSILICATE GLASS TECHNOLOGY

Borosilicate glass is the accepted waste form for the immobilization of liquid high-level radioactive waste, worldwide.<sup>5</sup> The glass compositions being utilized around the world are quite similar, all containing roughly one quarter waste by weight. Several different techniques for processing the glass are being utilized. In the U.S., the process being used at DWPF and West Valley, and planned for HWVP, is slurry-fed joule-heated ceramic melting. Glass frit, or glass-forming chemicals, needed to produce the glass product, are mixed with the waste to form the slurry feed. The slurry is continuous pumped into a ceramic-lined melter in which heat is added by passing electric current directly through the molten glass (joule heating). The major advantages of slurry-fed joule-heated ceramic melting are high throughput (design capacity of the DWPF melter is 40 liters of glass/hour) and low maintenance (no separate calciner, design life of the DWPF melter is at least two years). In 1987, a major U. S. radioactive campaign was completed which demonstrated the

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<sup>5</sup> An up-to-date summary of worldwide radioactive waste management can be found in National Briefing Summaries: Nuclear Fuel Cycle and Waste Management, PNL-6241, Pacific Northwest Laboratory, Richland, Washington, which is updated periodically. Table 2 (page OVR.7) in the most recent update (Revision 1, dated December, 1988) shows that glass is the reference high-level waste form in seven countries, glass and/or spent fuel are the reference waste forms in eight countries, and spent fuel is the reference waste form in three countries.

remote operation of the joule-heated ceramic melter. Brief descriptions of the progress in international implementation of borosilicate waste processing follow.

### France

As mentioned previously, France pioneered implementation of the immobilization of high-level radioactive waste in borosilicate glass. The AVM plant began immobilizing liquid high-level wastes at the Marcoule reprocessing facility in the south of France in 1978. Startup was smooth. Operation since has been relatively uneventful. Over 500 tons of borosilicate glass, contained in over 1,200 sealed stainless steel canisters, have been made and are in air-cooled storage at the Marcoule site.

France also has two reprocessing plants on its north coast at Cap de la Hague. Liquid high-level wastes are being converted to borosilicate glass. One plant, called R-7, began operation in 1979; a second plant, called T-7, is almost operational. These plants utilize the AVH process, which is basically the AVM process with minor modifications. Because the capacity of a single line is limited to about 5 liters of glass/hour, parallel lines are being used at the la Hague plants to increase capacity.

The French process differs from the U.S. process in several ways. The liquid waste, which is not neutralized and thus is acidic rather than alkaline, is calcined in a rotary calciner, then combined with glass frit and batch melted in an induction-heated metallic melter. The glass is drained from the melter into stainless steel canisters, which are welded shut and decontaminated before transfer to interim on-site storage. The metallic melter has a limited life and must be replaced periodically.

### Other European Countries

**Belgium-West Germany** - Conversion of the liquid high-level waste remaining at the decommissioned Eurochemic reprocessing plant at Mol, Belgium, to borosilicate glass began at the PAMELA plant in 1985. There are similar plans to convert the liquid high-level wastes from the WAK reprocessing pilot plant at Karlsruhe, Germany.

There have been aggressive research and development programs in Belgium and West Germany on the immobilization of liquid high-level wastes since the mid-1960s. These programs included investigation of several alternative waste forms and processes. Examples are the long investigation of phosphate glasses in West Germany and the development of the Vitromet process in Belgium. The Vitromet process, in which waste glass marbles are embedded in a lead matrix, was demonstrated with the fully radioactive Eurochemic wastes in the PAMELA plant. The Vitromet process is equally applicable to borosilicate or phosphate glass processing, but West Germany made a decision for borosilicate glass prior to the startup of PAMELA. The mainline effort at PAMELA is the operation of a joule-heated ceramic melter of German design for the conversion of the Eurochemic wastes to borosilicate glass; the same technology will be used for the WAK wastes.

United Kingdom - England has reprocessed nuclear fuel at the Windscale Works at Sellafield since 1964 and is constructing a new reprocessing facility, the Thorp Plant, at the same site. In addition, reprocessing of fast breeder reactor fuel has been done at Dounreay, in the north of Scotland. The liquid high-level wastes from all of these reprocessing plants will be converted to borosilicate glass.

The Atomic Energy Research Establishment was investigating methods of immobilizing radioactive wastes even before the Windscale Works were in operation, and operated a pilot plant at the Harwell laboratories for many years. A "rising level" glass process for incorporating liquid high-level waste in borosilicate glass was demonstrated at Harwell, but the ultimate decision was to use the only commercially available technology, the well-proven French AVM technology, at the Windscale Vitrification Plant (WVP). Construction of the WVP is complete, cold testing is well advanced, and hot operation is expected to begin in 1990. It is expected to take 12 years to vitrify the liquid high-level waste presently stored at Sellafield. The canisters of glass will be placed in air-cooled storage onsite.

Italy - The Italian Commission for Nuclear and Alternative Energy Sources operates two small reprocessing plants, the EUREX pilot plant at Saluggia in northern Italy, and the ITREC pilot plant at the Trisaia Center in southern Italy. There are plans to convert the liquid high-level waste at both sites to borosilicate glass using an in-pot vitrification process. The in-pot vitrification process has been developed in a nonradioactive pilot plant at the Trisaia Center and in hot cell tests at the Commission of the European Communities (CEC) Ispra laboratory in northern Italy.

#### Non-European Countries

India - India's first reprocessing plant began operation at Trombay in 1964. Two more reprocessing plants are now in operation, at Tarapur and Kalpakkam. A batch melting process has been developed in India for conversion of the liquid high-level wastes from these plants to borosilicate glass. The first Waste Immobilization Plant (WIP) began operation at Tarapur in 1985. Similar plants, but of advanced design, are scheduled to begin operation at Trombay in 1990, and at Kalpakkam in 1993.

Japan - The Japanese have a small reprocessing plant at Tokai that has operated intermittently for many years. A second reprocessing plant is scheduled to be in operation in about seven years. A pilot plant slurry-fed joule-heated ceramic melter has been constructed at Tokai. It is scheduled to begin processing the Tokai liquid high-level waste into borosilicate glass in 1991. There are similar plans to produce borosilicate glass at the second Japanese reprocessing plant.

The Japanese are also shipping spent fuel to France and England for reprocessing, with agreements that the high-level waste will be returned to Japan in the form of borosilicate glass. An air-cooled borosilicate glass storage facility is being constructed at Shimokita for interim storage of the returned

glass plus the glass produced by the second Japanese reprocessing plant; there will be a separate air-cooled interim storage facility at Horonobe for the borosilicate glass produced at Tokai.

Soviet Union - Russian reprocessing of nuclear fuel began in 1949 at a facility near Chelyabinsk (Kyshtym). This facility currently utilizes the same Purex process that is used in western nation reprocessing plants, thus the liquid high-level wastes are similar. After an extensive development effort, a large (500 L/h liquid feed rate) vitrification facility began operation in 1987. A liquid-fed joule-heated ceramic melter was used. About 160 metric tons of phosphate glass were made in two years before the facility was closed down due to failure of leads to the molybdenum electrodes. A two-stage borosilicate glass process is now being developed that will utilize a rotary calciner (similar to that used in the French AVM process) coupled to a joule-heated ceramic melter with different electrode design. There are also reports that, following the 1957 waste tank accident at Kyshtym, wastes were shipped to a special plant in the central Asian city of Krasnoyarsk, where they were to be stored and possibly vitrified, by an unknown process. The phosphate glass made at Kyshtym is in air-cooled storage on site.

### ALTERNATIVE WASTE FORM ACTIVITIES

There has been a decrease in alternative waste form investigation since the late 1970s and early 1980s, however some investigations are continuing. The most significant of these are reviewed briefly below. A recent book by Werner Lutze and Rodney C. Ewing (see Bibliography) contains a comprehensive description of alternative waste forms for liquid high-level radioactive waste.

#### Glass-Ceramic

Alternative waste form investigations in the U.S. in the eight years since the decision to make borosilicate glass in the DWPF have been limited to the development of a waste form for the Idaho National Engineering Laboratory (INEL) wastes, which will be described in this section, and some studies of phosphate glass, which will be described in the following section.

The INEL reprocessing facility, called the Idaho Chemical Processing Plant (ICPP), processes nuclear navy and test reactor fuels. The world's first plant for solidification of liquid high-level radioactive waste, by fluidized-bed calcination, began operation at the ICPP in 1963. The granular fluidized-bed calcine product is stored onsite in air-cooled stainless steel bins with an estimated 500-year design life.

A borosilicate glass formulation has been developed as a waste form for the ICPP calcine, but because of the large volume of expected future ICPP wastes, a waste form with higher waste loading is being sought. A glass-ceramic waste form appears promising. Glass-ceramics are a class of material consisting of a combination of crystalline phases and a glassy matrix; in the case of the conceptual ICPP waste form the volumetric ratio of crystalline phases to glass is approximately one to one. A high waste loading is achieved (>60% vs.

25-30%, which is usual in borosilicate glass) because waste constituents are contained in both the crystalline and glassy phases. For example, the high fluoride content of the ICPP waste is well-retained in the crystalline phase,  $\text{CaF}_2$ . The glass-ceramic waste form is produced by hot isostatic pressing (HIPing) a calcine: additive mixture. The process is still in a laboratory stage of development at ICPP. Processing of the ICPP wastes is not scheduled to begin until some time in the first decade of the twenty-first century.

### Phosphate Glass

Phosphate glass was first investigated as a high-level waste form at Brookhaven National Laboratory in the early 1960s. Eleven canisters of phosphate glass were made in the WSEP program at PNL, after which development work was abandoned in favor of borosilicate glass. West German investigators also studied phosphate glass for some time before making the decision for borosilicate glass. Although 160 metric tons of phosphate glass have been produced in the Soviet Union, the decision has apparently been made to convert to borosilicate glass as the preferred waste form.

The potential advantages of phosphate glass were that it could incorporate an even broader range of elements than could borosilicate glass, it permitted one-step processing (i.e., no calcining step was required), and it was apparently at least as insoluble as borosilicate glass under optimum conditions. The advantage of one-step processing disappeared with the advent of the slurry-fed ceramic melter, which made possible the manufacture of borosilicate glass in one step also. The insolubility of phosphate glass is much more dependent on glass composition and the absence of devitrification than is borosilicate glass. Thus, although the possibility of making phosphate glass that is as insoluble as borosilicate glass exists, process control requirements are much more restrictive. The lead-iron phosphate glass compositions studied at Oak Ridge National Laboratory in the mid-1980s (L. A. Boatner and co-workers) do not appear to overcome this disadvantage. Perhaps the biggest concern with phosphate glass is that there are no natural analogs. Silicate-based glasses have survived for long times in nature; no similar experience with phosphate-based glasses can be cited.

### Synroc

Synroc is the waste form that makes the most of the natural analog argument. Although natural silicate-based glasses, formed from volcanic activity or meteoric impact (tektites), have survived for long times in nature, the fact remains that glass is a metastable material. Geologic evidence is clear-cut. In the natural world, crystalline materials outlast glass. The Synroc concept is to identify a few naturally occurring minerals known to be resistant to weathering that, in combination, have crystalline structures that are able to accommodate all of the various fission products, actinides, and other waste constituents by atomic substitution in their lattice structures. By analogy, the "synthetic rocks" so formed should have stabilities similar to the related minerals. The Synroc that has been most studied consists of nuclear waste

substitutions in three main mineral phases, hollandite ( $\text{BaAl}_2\text{Ti}_6\text{O}_{16}$ ), perovskite ( $\text{CaTiO}_3$ ), and zirconalite ( $\text{CaZrTi}_2\text{O}_7$ ), together with minor and glassy intergranular phases.

Synroc development began in the late 1970s in Australia, where the concept originated (Ringwood, A. E., 1978, Safe Disposal of High Level Nuclear Reactor Wastes: A New Strategy, Australian National University Press, Canberra). DOE sponsored Synroc development work at Lawrence Livermore National Laboratory, Argonne National Laboratory and North Carolina State University as part of the National High-Level Waste Technology Program. During this period, a Synroc D composition was developed for the incorporation of Savannah River high-level wastes. Work on Synroc ceased in the U.S. following the 1982 decision to make borosilicate glass at the DWPF. A small amount of Synroc development persisted for a short while in Europe, but in the last few years Synroc development has been done only in Australia and Japan. The Japanese effort, being carried out at the Japan Atomic Energy Research Institute, is relatively minor.

The principal Synroc effort is in Australia. The Australian government has made development of the Synroc process a national priority. Australia has no high-level radioactive waste; they do supply major quantities of uranium to nuclear industries in other countries, and therefore feel a responsibility to assure that high-level radioactive wastes can be disposed of safely, wherever the wastes are generated. In their view, "Synroc is being developed as a second-generation high-level waste form with superior chemical durability to conventional glass waste forms," (D. E. Levins, in a presentation to the IAEA Research Coordination Meeting, June 5-9, 1989, Winnipeg, Canada).

The Australian Nuclear Science and Technology Organization (ANSTO) has constructed a nonradioactive Synroc Demonstration Plant at the Lucas Heights Research Laboratories. As of June 1989, 27 short-term campaigns had been conducted in the 10 kg/h plant. About 4,000 leach tests of Synroc products had been carried out at ANSTO to study the effects of process variables and leaching parameters. Many of the tests were carried out on products doped with fission products and actinides that were made in ANSTO's glove-box and hot cell facilities. The major findings are:

- Leach rates are usually slightly higher in deionized water than in groundwaters, but are relatively insensitive to the composition of the groundwater or its pH.
- The chemical durability of Synroc is relatively insensitive to changes in temperature. The overall leach rate of Synroc increases by a factor of 25 over the temperature range 45 to 250°C, whereas the rate for waste glasses typically increases by a factor of 250.
- Fabrication conditions can affect the chemical durability of Synroc. The more important parameters are hot pressing time and temperature, precursor type, homogeneity of mixing, and redox control during calcination and hot pressing.

- The initial leaching of Synroc is incongruent. Leachability decreases in the following order: Mo > Cs,I,Tc,Ba > Ca,Ru > Np > Zr,rare earths > Am,Cm,Pu.
- The leach rate of Ti (the principal Synroc phases are Ti-based minerals) is below detection limits.

The initial incongruent leaching is similar to that observed with borosilicate glass, and in fact may be principally due to interstitial glassy phases present in Synroc. The leaching advantage of Synroc is thought to be shown by the leach rate of titanium, which is assumed to be the matrix material in Synroc, and therefore the ultimate control of Synroc leachability when the leaching becomes congruent. The leach rate of titanium from Synroc is orders of magnitude lower than that of the matrix of borosilicate glass. It should be noted, however, that no leach tests have been conducted for sufficiently long periods to show that the leach rates of other Synroc constituents eventually become as low as that of titanium, i.e., congruent leaching of Synroc has not been demonstrated.

Although Synroc shows good potential for being a waste form with improved chemical durability, much development work remains to be done. Hot pressing on the scale required by the Synroc process has never been done in a remotely operated radioactive facility. This technology must be developed and demonstrated. The Synroc composition must be tailored to each individual waste composition; questions remain as to Synroc's ability to accommodate the variations and uncertainties that will be encountered in an actual high-level waste stream. Many years of additional development remain. Synroc is properly labeled a second-generation waste form.

#### SUMMARY AND CONCLUSIONS

Independent investigations conducted in many countries have led to borosilicate glass being the dominant choice as the waste form for immobilization of liquid high-level radioactive waste. Thirteen borosilicate glass vitrification facilities are in operation or under construction around the world. Decisions to convert high-level wastes to borosilicate glass were made at Savannah River and West Valley in 1982, and at Hanford in 1988 but only after an in-depth examination of the alternatives.

Borosilicate glass appears capable of meeting all waste form specifications that have appeared since 1982, including the NRC (1983) and EPA (1985) regulations for geologic disposal of high-level radioactive waste that provide limits for releases from the engineered barrier system, and to the accessible environment, respectively. The waste form is only one of many components affecting the releases at these points. Although much more study will be done before the repository license application is made, systems analyses made to date indicate that the performance of borosilicate glass in the system is satisfactory. Preliminary tests based on EPA RCRA classification methods show



that borosilicate glass containing high-level waste is characteristically non-hazardous. The major waste form criteria that are new since 1982 are DOE's WAPS. Borosilicate glass appears capable of meeting these specifications.

Because of the broad acceptance of borosilicate glass, there has been relatively little investigation of alternative waste forms in the last eight years. A glass-ceramic waste form is being developed for the high-level wastes at INEL. Some investigation of an improved phosphate glass formulation was conducted at the Oak Ridge National Laboratory, but at the same time the Soviet Union was apparently abandoning their manufacture of phosphate glass in favor of borosilicate glass. Development work on Synroc, generally believed to be the strongest competitor to borosilicate glass as a waste form, is being conducted only in Australia, augmented with a small laboratory-scale investigation in Japan. A nonradioactive Synroc pilot plant is operational in Australia, but no radioactive pilot-plant tests have been made, and no plans for such tests are known.

In conclusion, improvements in waste forms may be expected over time. That is the nature of technology. But for the HWVP, borosilicate glass remains the obvious waste form:

- Borosilicate glass was an appropriate choice (Synroc was considered in the selection),
- Borosilicate glass meets the new requirements,
- Synroc is potential second generation waste form, and
- This white paper finds no information that changes the relative waste form ranking (borosilicate glass over Synroc).

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This bibliography contains selected key documents published between 1982, the year that borosilicate glass was selected as the waste form for the DWPF, and the present. Some annotation is provided.

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The Evaluation and Selection of Candidate High-Level Waste Forms, 1982.  
DOE/TIC-11611, (Prepared by staff of the Savannah River Laboratory). 110  
pages.

A final ranking of the candidate waste forms that were evaluated in U.S. National High-Level Waste Technology Program. This major program, conducted from 1979 through 1981, started with 17 candidate waste forms for evaluation:

Borosilicate Glass (SRL,PNL)	High-Silica Glass (CUA,NPD)
Phosphate Glass (PNL,BNL)	Clay Ceramic (RHO,PNL)
Glass Ceramic (ICPP)	Tailored Ceramic (RI,PSU)
Synroc (LLNL,ANL,NCSU)	Titanate Ion Exchanger (SNL)
Stabilized Calcine (ICPP)	Pelletized Calcine (ICPP)
Normal Concrete (SNL,ORNL,PSU)	Hot-Pressed Concrete (PSU)
FUETAP Concrete (ORNL)	Matrix Forms (PNL,ANL)
Coated Sol-Gel Spheres (ORNL)	Cermet (ORNL)
Disc-Pelletized Coated Particles (PNL,BCL)	

(The letters in parentheses define laboratories at which research was conducted: ANL = Argonne National Laboratory, BCL = Battelle Columbus Laboratories, BNL = Brookhaven National Laboratory, CUA = Catholic University of America, ICPP = Idaho Chemical Processing Plant, LLNL = Lawrence Livermore National Laboratory, NCSU = North Carolina State University, NPD = NPD Nuclear Systems, Inc., ORNL = Oak Ridge National Laboratory, PNL = Pacific Northwest Laboratory, PSU = Pennsylvania State University, RHO = Rockwell Hanford Operations, RI = Rockwell International, SNL = Sandia National Laboratory, SRL = Savannah River Laboratory.)

As the screening process progressed, the number of candidate waste forms was narrowed to seven. These were ranked by a figure-of-merit technique that utilized input from DOE defense waste-sites and independent laboratories, peer review assessments, a product performance evaluation, and a processability analysis:

<u>Waste Form</u>	<u>Combined Figure-of-Merit</u>
Borosilicate Glass	75
Synroc	63
Tailored Ceramic	62
High-Silica Glass	57
FUETAP Concrete	55
Coated Particles	53
Glass Marbles in a Lead Matrix	48

The crystalline waste forms, Synroc and tailored ceramic, were ranked second to borosilicate glass because of their high product performance rating, even though their processability rating was lower than any of the other final candidate waste forms.

The Waste Acceptance Preliminary Specifications [OGR/B-8, 1986, for DWPF; DOE-RW-0136 (formerly OGR/B-9), January 1990, for WVDP].

The Waste Acceptance Preliminary Specifications (WAPS) are the keystone documents of the internal administrative process that the DOE has set up to ensure that immobilized high-level wastes will be acceptable at the geologic repository when a repository is ready to receive them in 2015. The separate WAPS that have been drafted for DWPF and WVDP contain the same specifications, listed below, and differ only in minor details (the major waste form specific specifications, 1.1 through 1.4, are identical in both WAPS):

#### Waste Form Specifications

- 1.1 Chemical Specification
- 1.2 Radionuclide Inventory Specification
- 1.3 Specification for Radionuclide Release Properties
- 1.4 Specification for Chemical and Phase Stability

#### Canister Specifications

- 2.1 Material Specification
- 2.2 Fabrication and Closure Specification
- 2.3 Identification and Labeling Specification

#### Canistered Waste Form Specifications

- 3.1 Free-Liquid Specification
- 3.2 Gas Specification
- 3.3 Specification for Explosiveness, Pyrophoricity, and Combustibility
- 3.4 Organic Materials Specification
- 3.5 Free-volume Specification
- 3.6 Specification for Removable Radioactive Contamination on External Surfaces
- 3.7 Heat Generation Specification
- 3.8 Specification for Maximum Dose Rates
- 3.9 Chemical Compatibility Specification
- 3.10 Subcriticality Specification
- 3.11 Specifications for Weight, Length, Diameter, and Overall Dimensions
- 3.12 Drop Test Specification
- 3.13 Handling Features Specification

#### Quality Assurance

The sites that will be sending borosilicate glass to the geologic repository are required to prepare 1) a Waste Compliance Plan (WCP) documenting the intended compliance with the WAPS provisions, and 2) a Waste Qualification Report (WQR) containing data that prove that the methodology described in the WCP will work. Both the WCP and WQR must be approved before the vitrification plants can begin processing radioactive waste. The WCP and WQR will also describe the production records that will accompany each canister of borosilicate glass delivered to the repository.

Reimus, M. A. H., G. F. Piepel, G. B. Mellinger and L. R. Bunnell. 1988. West Valley Glass Product Qualification Durability Studies, FY 1987-1988: Effects of Composition, Redox State, Thermal History, and Groundwater, PNL-6723, Pacific Northwest Laboratory, Richland, Washington. 109 pages.

A study using 40 different simulated waste glasses to determine the effect of glass composition on leachability. The approach was to prepare glasses with compositional variations around three reference glass compositions, yielding groups of 8 and 16 glasses in FY 1987, and another 16 glasses in FY 1988. All of the glasses were evaluated with a 7-day MCC-3 leach test; the 16 FY 1988 glasses were also evaluated with a MCC-1 leach test. The effects of variations in the following glass components were evaluated:  $\text{Al}_2\text{O}_3$ ,  $\text{B}_2\text{O}_3$ ,  $\text{BaO}+\text{CaO}+\text{MgO}$ ,  $\text{Fe}_2\text{O}_3$ ,  $\text{K}_2\text{O}+\text{Li}_2\text{O}+\text{Na}_2\text{O}$ ,  $\text{MnO}_2$ ,  $\text{P}_2\text{O}_5$ ,  $\text{SiO}_2$ ,  $\text{ThO}_2$ ,  $\text{UO}_2$ , and others (the remaining glass components which automatically varied as the preceding components were varied). Statistical analyses showed that the components having major effects were  $\text{K}_2\text{O}+\text{Li}_2\text{O}+\text{Na}_2\text{O}$ , which caused increased leachability with increasing concentration, and  $\text{SiO}_2$  and  $\text{Al}_2\text{O}_3$ , both of which caused decreased leachability with increasing concentration.

Specimens of one of the FY 1987 reference glasses were prepared with four different redox states. No statistical effect on leach rate was found. Specimens of the same glass composition were slow-cooled, producing differing amounts of cubic spinel-type  $[(\text{Fe},\text{Ni})(\text{Cr},\text{Fe})_2\text{O}_4]$  and  $\text{ThO}_2$  crystals, and, in some cases, hematite-type  $[(\text{Cr},\text{Fe})_2\text{O}_3]$  crystals and Ru inclusions. The leach rate (modified MCC-3 test) was increased a maximum of 40% by the devitrification. Specimens of the FY 1987 reference glass were leached in four different groundwaters. The leach rate (modified MCC-3 test) was up to five times lower than in deionized water. Microstructural examination of the 16 FY 1988 glasses (cooled normally) showed that the compositional variations did not produce marked changes in the non-glass phases in the glasses.

The average normalized boron release over 28 days for the 16 FY 1988 West Valley waste glasses tested by the MCC-1 procedure ranged from 0.28 to 0.68 g/m<sup>2</sup>/day, well below the 1.0 g/m<sup>2</sup>/day criterion in the WAPS. The authors caution that statistical factors, such as lab-to-lab and within-lab uncertainties, should not be ignored. When this is done, statistical analysis of the compositional model showed that some glasses at the extremes of the postulated composition range would not meet the 1.0 g/m<sup>2</sup>/day criterion at the 95% upper confidence limit.

A Study of the Isolation System for Geologic Disposal of Radioactive Wastes, 1983. National Academy Press, Washington, D.C. (commonly known as the "WISP" or "Pigford Panel" Report). 338 pages.

A two-year study conducted for DOE by an 11-member Waste Isolation Systems Panel of the Board on Radioactive Waste Management, under the National Research Council's Commission on Physical Sciences, Mathematics, and Resources. This is the major independent nuclear waste study conducted in the U.S. in the 1980s. It is of particular interest because it attempts to relate the waste form to the repository through performance assessment models, probably presaging the approach that will be used in licensing the repository. Thus, emphasis is on waste forms and geologic disposal; processing, interim storage, and transportation of waste are not considered. The scope of the report is shown by the chapter titles:

1. Executive Summary (15 pages)
2. The Charge to the Panel (4 pages)
3. The Geologic Waste-Disposal System (4 pages)
4. Waste Characteristics (20 pages)
5. The Waste Package (65 pages)
6. Conceptual Design of Repositories (37 pages)
7. Geologic, Hydrologic, and Geochemical Properties of Geologic Waste-Disposal Systems (66 pages)
8. Overall Performance Criterion for Geologic Waste Disposal (35 pages)
9. Performance Analysis of the Geologic Waste-Disposal System (54 pages)
10. Natural Analogs Relevant to Geologic Disposal (6 pages)

Waste forms are discussed in Chapter 5 on pages 51 through 83. The overall evaluation (p. 78) states: "Borosilicate glass is the appropriate choice for further testing and for use in current repository designs.", and, "Although glass may not be the ideal waste-form material, its specific properties are not critical and it will likely be adequate to meet our overall criterion for repository performance, based on the performance analyses in Chapter 9."



Lutze, W., and R. C. Ewing. 1988. Radioactive Waste Forms for the Future, North-Holland, New York. 778 pages.

A fine state-of-the-art review of waste forms for high level nuclear waste, with over 1,200 references. Nine waste forms are discussed in detail, silicate glasses, sintered glass, lead-iron phosphate glass, Synroc, tailored ceramics,  $TiO_2$ -ceramic matrix, glass-ceramics, monazite, and FUETAP concrete. Eight other waste forms are described briefly. The conclusion is that there are a large number of potential nuclear waste forms and, "This is an excellent situation, as the wide variety of nuclear waste streams and repository environments requires that waste forms be selected so as to optimize their long-term performance," (page 734). But there are only two waste forms, borosilicate glass and Synroc, for which pilot-scale or larger plants have been developed and built. (Note: The book was published before the Soviet experience with phosphate glass became general knowledge.) Full-scale radioactive production of borosilicate glass has been achieved; similar radioactive experience with Synroc is absent. The book ends with a description of the relative advantages of the two waste forms. Synroc has greater mechanical integrity and more stability in hydrothermal environments than borosilicate glass. There is no attempt to quantify the importance of these characteristics in terms of repository performance.

The book contains an 144-page chapter on silicate glasses, with over 500 references, that is the most comprehensive, up-to-date review of borosilicate glass properties available.

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**Figure 1**

Figure 1 consists of two panels, (a) and (b), showing the distribution of the number of clusters per site. Panel (a) shows the distribution for the first set of sites, and panel (b) shows the distribution for the second set of sites. Both panels have a y-axis labeled 'Number of clusters' ranging from 0 to 10 and an x-axis labeled 'Site' ranging from 1 to 10. The distributions are represented by bars.

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Winters, W. I., 1981 (June), *Effect of pH on the Destruction of Complexants with Ozone in Hanford Nuclear Waste*, RHO-SA-203, Rockwell Hanford Operations, Richland, Washington.

Wong, J. J., 1989 (October), *244-AR Conceptual Flowsheet for Processing of NCAW*, WHC-SE-WM-TI-396, Revision 0, Westinghouse Hanford Operations, Richland, Washington.

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Status or Availability: Public availability of document is being verified.

In January 1975 a program was established to evaluate technology for retrieval of solid and liquid wastes presently stored on the Hanford Reservation. The original program outline involved three tasks: 1) waste retrieval, 2) volume reduction and waste immobilization, and 3) equipment decommissioning. In March 1975 the scope of work was expanded to include a fourth task, development of concepts for an engineered storage facility.

"On the basis of thorough evaluation of existing information on the high level waste management alternatives, technologies, and criteria and on the basis of independent engineering analysis, the following recommendations were made:

- In retrieval operations, the reference design should be based on the mechanical in-tank material handling.
- In waste stabilization operations, the reference design should be based on the silicate melt processing alternative.
- In engineered storage operations, the reference design should be based on the modular bin concept.
- Finally, in equipment decommissioning operations, additional research and development should be conducted prior to the establishment of a reference design.

Augustine, C. A., 1989 (January), *Double-Shell Tank Waste Disposal Integration Plan*, WHC-EP-0229, Revision 1, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Bates, S. O., G. F. Piepel, and J. W. Johnston, 1989 (May), *Leach Testing of Simulated Hanford Waste Vitrification Plant Reference Glass HW-39*, PNL-6884, Prepared by Pacific Northwest Laboratory for Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

The document summarizes the work performed to investigate the viability of a leach testing methodology for the HWVP and provide glass

dissolution data for HWVP model determination and validation. Leach tests up to one year in duration were conducted on the reference glass HW-39-1. Some changes are recommended to the leach test methodology.

Buckley, L. L., and J. D. Kaser, 1983 (August), *Costs of Alternatives for the Disposal of Future PUREX High-Level Waste and Existing Waste in Double-Shell Tanks*, SD-WM-ES-019, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability: Document is being cleared concurrently with WHC-EP-0365.

The costs of selected alternatives for waste disposal were estimated. The cost effects of direct waste neutralization and byproduct removal were estimated. Costs for major waste processing and disposal alternatives were identified. The conclusions drawn are:

- All viable alternatives require operation of B Plant.
- Removal of cesium is cost-effective from a waste management standpoint.
- The cost of purification and encapsulation of cesium for beneficial use is comparatively small.
- Removal of strontium is not cost effective from a waste management standpoint.
- Purification and encapsulation costs for strontium are substantial.
- Relative to the Savannah River approach, use of B Plant to remove cesium allows utilization of existing Hanford capabilities and experience with no apparent cost penalty.

DeFigh-Price, C., and B. A. Higley, 1985 (June), *Project Engineering Bases for Treatment of Double-Shell Tank Wastes: Process and Facilities Options*, SD-WM-ES-065, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This document contains cost estimates of alternative waste management plans and schedules prepared by Science Applications, Inc., Richland, for Rockwell Hanford Operations as 'Project Engineering Bases for Treatment of Double-Shell Tank Wastes' Process flowsheets on which the cost estimates are based, are presented in SD-WM-ES-025, 'Preliminary Process Flowsheets for Treatment of Hanford Defense Liquid Wastes.' These reports are summarized in SD-WM-ES-023, 'Evaluation of Process and Facility Options for Treatment of Double-Shell Tank Wastes.'

DOE, 1981 (July), *The Evaluation and Review of Alternative Waste Forms for Immobilization of High-Level Radioactive Wastes*, DOE/TIC-11472, U.S. Department of Energy, Washington, D.C.

Status or Availability (Checked): Publicly available document.

The document presents the relative merits and potential of eight alternative waste forms for disposal of high-level radioactive waste. The eight waste forms were selected from 15 previously evaluated. A Peer Review Panel composed of eight scientists and engineers representing independent, non-DOE laboratories from industry, government, and universities and disciplines of material science, ceramics, glass, metallurgy, and geology did the review. The waste forms were ranked as follows: borosilicate glass, SYNROC, porous glass matrix, tailored ceramics, pyrolytic C and SiC coated particles, FUETAP concrete, metal matrices, and plasma spray coatings.

DOE, 1981 (August), *The Evaluation and Selection of Candidate High-Level Waste Forms*, DOE/TIC-11611, U.S. Department of Energy, Savannah River Operations Office, Aiken, South Carolina.

Status or Availability (Checked): Publicly available document.

Seven candidate waste forms developed under the direction of the DOE's National High-Level Waste Technology Program, were evaluated as potential media for the immobilization and geologic disposal of high-level nuclear wastes. The evaluation combined preliminary waste form evaluations conducted at the DOE defense waste sites and independent laboratories, peer review assessments, a product performance evaluation, and a processability analysis. Based on the combined results of these four inputs, two of the seven forms, borosilicate glass and a titanate based ceramic, SYNROC, were selected as the reference and alternative forms for continued development and evaluation in the National HLW Program. Both the glass and ceramic forms were depicted as viable candidates for use at each of the DOE defense waste-sites. This report describes the waste form screening process, and discusses each of the four major input considered in the selection of the two forms.

DOE, 1982 (July), *Environmental Assessment, Waste Form Selection for SRP High-Level Waste*, DOE/EA-0179, U.S. Department of Energy, Washington, D.C.

Status or Availability (Checked): Publicly available document.

This document presents borosilicate glass as the proposed waste form for disposal of SRP HLW, and crystalline ceramic as the leading alternative. A description of the properties, processing requirements, and development requirements for the proposed and alternate waste forms is provided. An assessment of the environmental consequences of the use of these two waste forms is presented. The document also lists 17 candidate waste forms that were considered for geologic disposal and describes the screening process by which borosilicate glass and crystalline ceramic were selected for further development.

DOE, 1987 (December), *Final Environmental Impact Statement, Disposal of*

*Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington, DOE/EIS-0113, 5 volumes, U.S. Department of Energy, Washington, D.C.*

Status or Availability (Checked): Publicly available document.

The purpose of this Environmental Impact Statement (EIS) is to provide environmental input into the selection and implementation of final disposal actions for high-level, transuranic and tank wastes located at Hanford Site, Richland, Washington, and into the construction, operation and decommissioning of waste treatment facilities that may be required in implementing waste disposal alternatives. Specifically evaluated are a Hanford Waste Vitrification Plant, Transportable Grout Facility, and a Waste Receiving and Packaging Facility. Also an evaluation is presented to assist in determining whether any additional action should be taken in terms of long-term environmental protection for waste that was disposed of at Hanford prior to 1970 as low-level waste (before the transuranic waste category was established by the Atomic Energy Commission but which might fall into that category if generated today).

The following alternatives are considered in this EIS: 1) in-place stabilization and disposal, where waste is left in place but is isolated by protective and natural barriers; 2) geologic disposal, where most of the waste (by activity and to the extent practicable is exhumed, treated, segregated, packaged and disposed of in a deep geologic repository; waste classified as high-level would be disposed of in a commercial repository developed pursuant to the Nuclear Waste Policy Act; transuranic waste would be disposed of in the Waste Isolation Pilot Plant near Carlsbad, New Mexico; 3) a reference alternative, where some classes of waste are disposed of in geologic repositories and other classes of waste are disposed of by in-place stabilization and disposal; 4) the preferred alternative, in which double-shell tank wastes, strontium and cesium capsules, and retrievably stored TRU wastes are disposed of according to the reference alternative, and in which double-shell tank wastes, strontium and cesium capsules, and retrievably stored TRU wastes are disposed of according to the reference alternative, and in which decisions are deferred on disposal of single-shell tank wastes and on further remedial action for TRU-contaminated soil sites and pre-1970 buried suspect TRU-contaminated solid wastes (except the 618-11 site) until additional information is obtained on waste characterization, retrieval methods, and performance of new-surface disposal systems; and 5) a no disposal action alternative (continued storage).

DOE, 1989 (November), *Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Revision 5, Prepared by Oak Ridge National Laboratory, Oak Ridge, Tennessee, for U.S. Department of Energy, Headquarters, Washington, D.C.

Status or Availability (Checked): Publicly available document.

DOE-RL, 1983, Hanford Waste Management Plan, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Dosch, R. G., 1978 (June), *The Use of Titanates in Decontamination of Defense Waste*, SAND78-0710, Prepared by Sandia Laboratories, Albuquerque, New Mexico, for Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Sodium titanate, an inorganic ion exchange material, has been evaluated for use in a process to remove strontium from Defense Waste or other sodium, caustic solutions. Distribution coefficients on the order of  $10^5$  were observed at sub part per million concentration of Sr, and the effects of other cation impurities and complexants in the waste were investigated.

The preparation and general chemical properties of the exchange material are discussed. This information was used in developing a commercial source which has since supplied a 200 kg batch of the material for evaluation.

In column ion exchange experiments with  $^{85}\text{Sr}$ -doped simulated waste, decontamination factors of 500 or greater were observed in the first 2000 to 3500 bed volumes of effluent, depending on the impurities in the simulant. A -40 to +130 mesh range of sodium titanate powder was used as the baseline material, but a study to produce alternate forms of the titanate was carried in parallel. This has resulted in two materials which appear promising with respect to both simplification of handling and chemical properties. One of the materials is an agglomerated form of the titanate formed by extrusion pelletizing using water as a binder, and the second is a macroreticular organic anion resin which was loaded with 30 to 40% (by weight) of sodium titanate. The results of initial testing of these materials are discussed.

Dunson, Jr., J. B., A. M. Eisenberg, R. L. Schuyler, III, H. G. Haight, Jr., V. E. Mello, T. H. Gould, Jr., J. L. Butler, and J. B. Pickett, 1982 (March), *Assessment of Processes, Facilities, and Costs for Alternative Solid Forms for Immobilization of SRP Defense Waste*, DP-1625, E. I. duPont, de Nemours & Company, Aiken, South Carolina.

Status or Availability (Checked): Publicly available document.

The document presents a quantitative merit evaluation of the relative difficulty of remote processing of SRP HLW for seven waste forms. The borosilicate glass process is rated as the simplest. The other processes evaluated in order of increasing complexity were: FUETAP concrete, glass marbles in a lead matrix, high-silica glass, crystalline ceramic, and coated ceramic particles. Cost appraisals are summarized for the borosilicate glass, high-silica glass, and ceramic waste form processing facilities.

Ecology, EPA, and DOE, 1989 (May), Hanford Federal Facility Agreement and Consent Order, Washington State Department of Ecology, Olympia, Washington; U.S. Environmental Protection Agency, Region 10, Seattle, Washington; and U.S. Department of Energy, Richland, Washington.

Status or Availability (Checked): Publicly available document.

ERDA, 1975 (December), *Final Environmental Statement, Waste Management Operations, Hanford Reservation, Richland, Washington*, ERDA-1538, 2 volumes, U.S. Environmental Research and Development Administration, Richland, Washington.

Status or Availability (Checked): Publicly Available Document.

This Final Environmental Statement has been prepared toward compliance with the National Environmental Policy Act to assess the environmental impact of continuing the ERDA's waste management operations at the Hanford Reservation in Benton County, Washington.

Alternatives to current waste management operations are presented for radioactive and non-radioactive liquid, solid, and gaseous waste. Alternatives to the current waste management operations discussed include both the additional treatment of waste streams, discontinuance of solidification to salt cake and the reduction of waste generation by curtailment of operations at the site. Ceasing waste management operations is not considered due to the continuing need to manage existing radioactive waste at Hanford.

ERDA, 1977 (September), *Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Hanford Reservations, Richland, Washington*, ERDA 77-44, U.S. Energy Research and Development Administration, Washington, D.C.

Status or Availability (Checked): Publicly available document.

The objective of this document is to provide information on alternatives that are being considered for the long-term management of defense high-level radioactive waste stored at Hanford in underground tanks and in stainless steel-lined concrete basins. For purposes of basic programmatic decisions making, four major alternatives based on disposal location are considered:

- existing waste tanks
- onsite engineered surface facilities
- onsite geologic repository
- offsite geologic repository.

The four major disposal alternatives are expanded into 27 alternative plans by considering:



- Variations in the final form of the high-level fraction (with radionuclide removal) to include glass, concrete, and powder.
- Variations in the final form of the dehydrated waste product to include glass, calcined clay, and powder.
- Variations in the treatment and handling of encapsulated waste to include packaging of capsules in canisters and conversion of the strontium fluoride and cesium chloride to glass.

A description of the technology, a preliminary risk assessment, and preliminary cost estimates for each of these 27 plans are presented.

ERDA, 1977 (May), *Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Savannah River Plant, Aiken, South Carolina*, ERDA 77-42/1, 2 volumes, U.S. Energy Research Administration, Richland, Washington.

Status or Availability (Checked): Publicly available document.

This document was prepared to provide other Government agencies and the public information on possible alternatives which will be considered for the long-term management of Savannah River Plant (SRP) high-level nuclear waste. It describes a number of alternative plans for long-term management or disposal of the high-level nuclear wastes now stored in tanks at the Savannah River Plant near Aiken, South Carolina.

The Savannah River Plant operations produce high-level radioactive waste in the chemical processing of fuel and target elements after irradiation in the SRP nuclear reactors. This waste is stored as an alkaline liquid with a precipitated sludge until the decay heating has abated appreciably. The supernatant liquid is then converted to salt cake to reduce volume and mobility.

The purpose of the site-specific document is to describe the different alternatives along with their probable relative costs, risks, and uncertainties. A secondary purpose is to raise the issue of methodology for decision making in nuclear waste management. The document does not attempt to arrive at any recommendations.

Implementation costs and risk costs are calculated in the text for 23 alternative plans for long-range management and isolation of the SRP high-level radioactive waste. For purposes of basic programmatic decision making, these 23 plans can be grouped into four main classes (Figure II-1):

1. Convert the waste to a highly leach-resistant form, such as canned glass cylinders, and ship it offsite to a Federal repository.
2. Convert the waste to a highly leach-resistant form, and store the waste in an engineered surface facility at SRP.

3. Reconstitute the waste to a slurry, and dispose of it in a bedrock cavern under the SRP site.
4. Continue storage in tanks with the waste as salt cake and sludge."

Freeberg, R. D., 1989 "Tri-Party Agreement Milestone M-04-01" (External Letter 8905292 to P. T. Day, U.S. Environmental Protection Agency, Region 10, and R. G. Stanley, Washington State Department of Ecology, December 21, 1989), U.S. Department of Energy, Richland Operations Office, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Hammitt, A. P., and W. W. Schulz, 1978 (September), *Hot Cell Facility and Equipment for Test of the Hanford Radionuclide Removal Process*, RHO-SA-52, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Bench-scale tests of ion exchange, precipitation, and other separation processes for removal of  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , actinides, and various other radionuclides from the water-soluble portion of the Hanford Defense Wastes have been successfully completed. This paper describes the hot cell and associated equipment to be used in further, scaled-up development and demonstration of the Hanford Radionuclide Removal process.

Hanlon, B. M., 1990 (June), *Tank Farm Surveillance and Waste Status Summary Report for April 1990*, WHC-EP-0182-25, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Hannum, W. H., 1983 (January), *Analysis of the Terminal Waste Form Selection for the West Valley Demonstration Project*, DOE/NE/44139-T3, Prepared by West Valley Nuclear Services Company, West Valley, New York, for the U.S. Department of Energy, Washington D.C.

Status or Availability (Checked): Publicly available document.

This document summarizes the environmental considerations associated with the selection of borosilicate glass as the waste form for the disposal of approximately 560,000 gallons of liquid HLW stored at West Valley. Product performance criteria discussed include leach resistance, thermal stability, mechanical stability, radiation stability, mechanical strength, impact resistance, fire resistance, and waste loading.

Higley, B. A., 1984 (April), *Preliminary Process Flowsheets for Treatment of Hanford Defense Liquid Wastes*, SD-WM-ES-025, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This document compiles the preliminary process flowsheets which were prepared for use in estimating the costs of alternative waste management schedules and plans. Details of the cost estimates are presented in SAI-84-3013 "Project Engineering Bases for Treatment of Double Shell Tank Wastes: Process and Facility Options" while the "Evaluation of Process and Facility Options for Treatment of Double Shell Tank Wastes" is summarized in SD-WM-ES-023. These flowsheets account for the principal operations and capabilities required to pretreat and dispose of the waste as glass and grout. Eight flowsheets have been developed which describe the pretreatment, vitrification and transportable grout process.

They are:

- Waste removal and transfer
- Complexant destruction by ozonization
- Solids removal and washing
- Cesium removal by ion exchange
- Transuranic contaminant separation from cladding removal waste
- Low-level waste concentration
- Vitrification
- Low-level waste disposal by grout.

Higley, B. A., 1988 (January), *Impact of Alternative Single-Shell Tank Waste Retrieval and Pretreatment Scenarios on the Hanford Waste Vitrification Plant*, SD-WM-TA-014, Revision 0, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This study examines the practicality of using the Hanford Waste Vitrification Plant (HWVP) to vitrify retrieved single-shell tank (SST) wastes. Four waste pretreatment alternatives for the retrieval of SST waste are considered. Three of the alternatives assume application of TRUEX process technology. The current HWVP design will allow installation of a 100 kg/hr glass melter without major modifications; this melter would be installed in the event that SST retrieval is required.

It is concluded that the HWVP as currently designed to accommodate a 100 kg/hr glass melter, is adequate to vitrify waste from a SST retrieval mission when TRUEX process technology is used in the waste pretreatment process. It is estimated that the use of TRUEX process technology could save 5,100 to 9,000 million dollars, depending on variations in the pretreatment process, relative to a base case in which washed sludge is vitrified.

Higley, B. A., and W. W. Schulz, 1988 (August), *Evaluation of Selected Alternatives for Processing Retrieved Hanford Single-Shell Tank Wastes*, WHC-EP-0191 DRAFT, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Costs and various other impacts and features associated with the retrieval, processing, and immobilization (grout and/or vitrification) of various amounts of single-shell tank (SST) wastes were evaluated. Three different waste retrieval scenarios were considered: retrieval of wastes from 149, 75 and 12 SSTs. For each retrieval scenario, the effect of two processing treatments [simple sludge washing and sludge washing coupled with the Transuranic Extraction (TRUEX) process] on the final amount of disposed waste and on overall disposal costs was determined.

Cost savings from sludge washing coupled with the TRUEX process, when compared to simple sludge washing, range from 700 million to about 9 billion dollars depending on the number of SSTs involved (Table 1). Both capital and expense dollar savings can be realized by implementation of the TRUEX process. Substantial reductions in expenditures for Hanford Waste Vitrification Plant (HWVP) operation and for fees for geologic disposal of vitrified SST waste contribute to savings in expense dollars. Savings in capital funds result from eliminating the need to construct additional HWVPs.

Without additional processing steps, application of the TRUEX process to acid dissolved, water-washed sludge could route  $^{90}\text{Sr}$  and uranium to the grout product. In all cases,  $^{137}\text{Cs}$  will, unless removed, be incorporated in grout for subsurface disposal. Comprehensive performance assessments must be performed to fully evaluate environmental impacts of subsurface disposal in engineered vaults, of all or part of the SST inventory of  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , uranium, and other radionuclides.

If necessary, as demonstrated by the results of performance assessments, well-known ion exchange technology can be used to remove  $^{137}\text{Cs}$  from alkaline SST solutions; the concentrated  $^{137}\text{Cs}$  fraction would constitute part of the feed to the HWVP. Precipitation, ion exchange, and solvent extraction processes all appear potentially applicable to removal of  $^{90}\text{Sr}$  from acidic TRUEX process raffinate. The concentrated  $^{90}\text{Sr}$  fraction could also be vitrified in the HWVP.

Hill, O. F., 1970 (August), *Salt Conversion Into Metal (SCIM)*, ARH-1810, Atlantic Richfield Hanford Company, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

Hodgson, K. M., 1979 (December), *Status of Solids/Liquids Separation Development for Separation and Concentration of Hanford High-Level*

Defense Waste, RHO-CD-846, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The separation and concentration process uses water washing and ion exchange to accomplish a chemical separation of the radionuclides from industrial chemicals. The resulting small quantity of highly radioactive wastes is then ready to be incorporated into an immobile waste form. The majority of the radionuclides are contained in the insoluble chemical fraction (sludge) which precipitated from the original liquid waste stored in the underground tanks. The purpose of the sludge washing process is to reduce the volume of the radioactive material that must be immobilized by removing water soluble industrial chemicals from the insoluble sludge and to reduce the concentration of soluble chemicals that tend to degrade the immobile waste forms. This volume reduction results in a substantial cost savings in the storage and disposal process. This savings is realized for all waste forms. Sodium ions also degrade such waste forms as glass, ceramics, bitumen, and concrete. Sulfate, which is removed by washing, degrades glass and ceramic waste forms. The reduction of nitrate by washing is desirable for high-temperature waste forms. The purpose of this document is to report the status of the sludge washing and solids/liquids separation process development activity.

The sludge washing and solids/liquids separation development work has shown that the sodium and sulfate ion concentrations can be reduced to acceptable levels and that solids/liquids separation can be obtained with some of the methods tested. However, further development is necessary.

Gravity thickening is a potential solids/liquids separations method. Separations was obtained when the original synthetic sludge contained 550 ppm of an anionic polyelectrolyte. Additional gravity settling test are required with actual sludges to determine if they will perform the same as the synthetic sludges.

In order to select a solids/liquids separation method, additional studies need to be performed on the many types of separation methods available. This can be accomplished through onsite testing and offsite vendor tests.

Jantzen, C. M., 1988 (November), *Glass Compositions and Frit Formulations Developed for DWPF*, DPST-88-952, Savannah River Laboratory, Savannah River, Georgia.

Status or Availability (Checked): Public availability of document is being verified.

The document describes a family of glass compositions and frit formulations that have been developed for DWPF. The history of the

development work is summarized. The result is a recommendation of a frit for initial DWPF operations which when vitrified with the DWPF waste, over its composition range, will meet the processing and repository acceptance requirements.

Kaser, J. D., 1985 (September), *Removal of Transuranic (TRU) Elements from PUREX Current Acid Waste (CAW) and Neutralized Current Acid Waste*, SD-WM-TA-011, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This study assesses transuranic (TRU) element separation from PUREX neutralized current acid waste (NCAW) generated through 1996. Transuranic element separation reduces the cost of vitrification by reducing the volume of glass produced. In addition to solid-liquid separation, the only promising process for TRU element removal is the TRUEX solvent extraction process. Three facilities were considered for treating the waste by the TRUEX process: 1) B Plant, 2) head end of the PUREX Plant, and 3) The HWVP with added cell space.

The lowest cost alternative is to remove TRU from the waste in the head end of PUREX after implementation of PUREX Facility Modification (PFM) shear/leach decladding. The life cycle cost for this option is estimated at \$173 million, which is \$100 million less than the currently planned alternative of vitrifying all NCAW sludge. Implementation of TRUEX in PUREX would require the loss of a dedicated spare aging waste tank from 1990 to 1993. However, one million gallons of spare aging waste storage space will still be available.

The volume of grout is increased from 3.9 million gallons for the current alternative to 6.9 million gallons for TRUEX process operation, and most of the radiostrontium will end up in the grout rather than in the glass. The maximum concentration of <sup>90</sup>Sr, which can be tolerated in grout disposed of near surface, needs to be determined. If <sup>90</sup>Sr separation is required, the best method of separation must be identified and the cost of <sup>90</sup>Sr separation must be estimated.

Kaser, J. D., and B. A. Higley, 1984 (August), *Alternatives for Treatment of Neutralized Plutonium Finishing Plant Liquid Wastes for Disposal*, SD-WM-ES-038, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The following six alternatives for treatment and disposal of liquid TRU waste from the Plutonium Finishing Plant (PFP) are evaluated:

- Continue current PFP operating mode. Vitrify all sludge for geologic disposal.

- Continue current PFP operating mode. Convert all sludge to grout for geologic disposal.
- Continue current PFP operating mode. Treat all sludge in B Plant for TRU separation and Plutonium recovery.
- Continue current PFP operating mode to 1991. Vitrify pre 1991 sludge. Beginning in 1991 treat waste in PFP for TRU separation and plutonium recovery.
- Continue current PFP operating mode to 1991. Treat all waste for TRU separation and plutonium recovery. Pre 1991 waste is treated in B Plant and post 1991 waste at PFP.
- Initiate TRU separation and Pu recovery at PFP as soon as possible. Treat prior generated waste in B Plant for TRU separation and Plutonium recovery.

The last alternative is the least expensive and recovers up to 410 kg of plutonium, while the first alternative is the most expensive and recovers no plutonium.

Kaser, J. D., B. A. Higley, and M. J. Kupfer, 1983 (June), *Alternatives for Disposal of Hanford Liquids & Sludges Which May Not be Suitable for In Situ Disposal*, SD-WM-ES-012, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The cost of selected options for solidifying and disposing of radioactive waste liquids and sludges are compared. Glass and concrete were the two waste forms considered for geologic disposal. The cost impacts on waste disposal of cesium, strontium and TRU element removal were estimated. By-product and TRU separation costs are not included.

The major findings of this study are:

- Packaging & repository disposal are the most expensive process elements.
- Removal of radiocesium greatly reduces the cost of disposing of the non-TRU portion of the waste.
- Removal of TRU can result in large savings in packaging and disposal costs.
- Repository disposal of glass may be less expensive than repository disposal of concrete."

KEH, 1977 (September), *Final Report, Hanford Defense High-level Waste Management Studies*, 77-09-RE, Prepared for the U.S. Energy Research and Development Administration by Kaiser Engineers Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

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This document was prepared to provide information on possible alternatives that will be considered for the long-term management of high-level radioactive nuclear waste accumulated as part of the national defense effort at the Hanford Reservation near Richland, Washington. It describes a number of alternatives for retrieval, treatment, and long-term storage of the raw wastes now stored in underground tanks and the treated waste stored in water basins at Hanford. The descriptions include implementation technology, a safety assessment, and preliminary cost estimates. The cost estimates, although useful to compare alternatives, are not of budget quality.

Kaiser Engineers prepared this document in conjunction with a report published by the U. S. Energy Research and Development Administration titled "Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, Hanford Reservation, Richland Washington" ERDA document ERDA-77-44. The Kaiser Engineers report is intended to support this ERDA document by providing more detailed information concerning the waste management alternatives discussed in ERDA-77-44.

This document does not take into account either social and public policy issues or the environmental impacts of the alternatives discussed. Instead, the document presents information that is possible to quantify concerning the technology, safety, and costs of waste management alternatives to provide a preliminary basis for discussion and judgement in future decision-making. No selection or recommendation of an alternative for implementation is made in this document. Information contained in this document will be considered in the preparation of a programmatic environmental impact statement and in the selection of waste forms) and storage mode(s) for long-term management of these wastes."

Klem, M. J., J. F. Fletcher, C. E. Golberg, R. D. Gibby, K. A. Giese, F. A. Ruck, J. C. Sonnichsen, D. D. Wanner, N. R. Wing, and K. A. Woodworth, 1990 (June), *Technology Program Plan for Closure of the Single-Shell Tank Operable Units*, WHC-EP-0288, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

This Technology Program Plan for Closure of the Single-Shell Tank Operable Units (TPP) provides documentation of the required technology, resources, equipment, program funding, and plans for closure of the six single-shell tank (SST) operable units (OU). The SST OUs comprise treatment, storage, and disposal units (wastes, tanks, and soil contaminated by leaks) and past practice units (ancillary units and soil contaminated by spills). These units are regulated under the Resource Conservation and Recovery Act Section 3005 (e) (interim status permit authority) and Section 3004 (u) (past practices).

A system engineering approach is being used as a management tool to assist in reaching a final disposal decision for the SST OUs. The systems approach is a structured process to define and solve a problem.



It is useful for large programs that involve multiple scientific and engineering disciplines and span long time periods. The systems approach ensures that development activities are conducted in an integrated, efficient, thorough, logical, defensible, auditable, and verifiable manner. It will allow the U.S. Department of Energy to meet Hanford Federal Facility Agreement and Consent Order milestones and develop the technology required for the supplemental environmental impact statement for SST waste. The systems engineering for closure of the six single-shell tank OUs is broken down into nine function areas.

These nine functions represent a set of actions that will be permanent throughout the development and implementing phases. This breakdown becomes the framework for planning as the program transits from development to implementation. The functions are divided into main elements or subfunctions and related tasks to provide more detail. Descriptions, special assumptions and constraints, projected costs, and schedules were developed to quantify the requirements and provide a baseline for future planning.

It does not appear economically attractive to vitrify the large amounts (up to 1400 metric tons in 149 tanks) of uranium isolated by TRUEX process operation with dissolved water-washed SST sludge. If not acceptable for subsurface disposal in grout form, the TRUEX process uranium product could be purified further (e.g., by a tri-*n*-butyl phosphate extraction process), calcined to  $UO_3$ , and stored until it can be satisfactorily used in some part of the U.S. Department of Energy (DOE) nuclear fuel cycle.

Kupfer, M. J., 1987 (March), *Costs and Impacts of Retrieval and Processing of Wastes from Selected Single-Shell Tanks*, RHO-WM-EV-17P, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

This report determines the costs of retrieval and processing of wastes from selected Hanford Site underground single-shell tanks and the impacts of retrieval on waste pretreatment operations, grout, and glass production. The assumptions and methods used for determining the costs are consistent with those used for costing the disposal alternatives described in the Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the Hanford Defense Waste Environmental Impact Statement (Rockwell 1985).

Retrieval options were chosen based on projected transuranic element inventory and transuranic element concentration in wastes in single-shell tanks. Retrieval options range from as few as one single-shell tank to as many as 116 tanks. The case of retrieving all 149 single-shell tanks is covered in Rockwell 1985.

Waste is assumed to be removed from single-shell tanks using mechanical retrieval equipment. The feed pretreatment steps include dissolution

of the salt cake and separation of the sludge and dissolved salt cake by centrifugation. The sludge is washed with water to assure dissolution of soluble salts. Organic complexants are destroyed in waste from certain single-shell tanks by ozonization of the dissolved salt cake. The dissolved salt cake is converted to a cementitious grout form for near-surface disposal in concrete vaults. The washed sludge is converted to borosilicate glass for disposal in a deep geologic repository.

Disposal of single-shell tank and double-shell tank waste (both existing and future) is estimated to cost 1.7 billion fiscal year (FY) 1983 dollars for the Reference Alternative disposal option (Rockwell 1985). The Transportable Grout Facility will be able to process single-shell tank waste for selective-retrieval options involving retrieval of waste from up to 61 tanks, as well as the existing and future double-shell tank waste. An additional facility is required if 61 or more tanks are retrieved. New feed tanks and pipelines would be required to accommodate feed from the single-shell tanks. The glass melter will be capable of vitrifying waste from only about five single-shell tanks, in addition to vitrification of the existing and future waste from double-shell tanks. However, this is based on the conservative assumed throughput of only 30 kg of glass per hour and a maximum campaign time of 18 yr (used in Rockwell 1985). (In the present conceptual design both the melter throughput and campaign time can be expanded.)

Kupfer, M. J., 1989 (July 31), *Evaluation of Costs for Selected Retrieval and Processing of Wastes from Single-Shell Tanks*, SD-WM-TI-226, Revision 0, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The costs of retrieval and processing of wastes from selected single-shell tanks (SSTs) and the impacts of retrieval on waste pretreatment operations, grout, and glass production were determined. The assumptions and methods used for determining costs for the selective retrieval options were consistent with those used for costing the disposal alternatives described in the Hanford Defense Waste - Environmental Impact Statement Engineering Data Packages (HDW-EIS EDP). The transuranic (TRU) inventory and TRU concentrations in SSTs were used as the basis for choosing candidate tanks for retrieval.

The following impacts on processing operations were identified:

- B Plant can handle dissolution and sludge washing of waste from 8-14 SSTs based on a maximum operating campaign of 18 years.
- The Transportable Grout Facility can process the maximum volume of SST waste envisioned for the partial retrieval scenarios, as well as the existing and future double-shell tank (DST) waste. However,

new feed tanks and pipelines would be required to accommodate feed from the SSTs.

- The glass melter will be capable of vitrifying waste from only about five SSTs in addition to the existing and future waste from DSTs. However, this is based on the conservative HDW-EIS EDP assumed throughput of only 30 kg glass/hr and a maximum campaign time of 18 years.

Kupfer, M. J., A. L. Boldt, and J. L. Buelte, 1989 (September), *Process and Facility Options for Pretreatment of Hanford Tank Waste*, SD-WM-TA-015, Revision 0, Westinghouse Hanford Company, Richland, Washington.

Status or Availability: Document is being cleared concurrently with WHC-EP-0365.

The subject report provides an assessment of process and facility options for treating Hanford Site tank waste for immobilization and final disposal. Currently known options for treatment and immobilization of double-shell tank (DST) wastes, new and existing facilities for performing the processing operations, and the timing and capacity of needed feed pretreatment facilities are evaluated. The lower cost processing and facility options that are of reasonable technical certainty are identified. A preferred option is identified that can result in a potential waste disposal program savings of \$500 million. The preferred option involves water washing of neutralized current acid waste (NCAW) sludge in a DST or in the 244-AR Vault (rather than in B Plant), and accelerating implementation of the transuranic extraction (TRUEX) process at B Plant for treatment of follow-on DST wastes. Increasing the vitrification capacity for DST wastes from 45 kg/h to 100 kg/h is also recommended for the preferred option.

Major issues pertaining to both waste processing and facility options, and appropriate development requirements to resolve these issues are identified.

This report provides information that was developed and presented in draft form in fiscal year (FY) 1988. Several follow-on studies have since been performed that addressed key items and recommendations made in this report. The report "Assessment of Double-Shell Tank Waste Pretreatment Options," (Sec 3.3.2) summarizes this information. The conclusions and recommendations in this report has not been updated to incorporate any changes to major assumptions, e.g., those associated with operational schedules, milestones, and costs. Issuance of this report in final form provides detailed background information and bases that support the more recent studies.

Neutralized current acid waste (NCAW) sludge washing will be performed at the 224-AR Vault. This document contains a revised flowsheet description for NCAW pretreatment at AR Vault and B Plant. Upgrades to both facilities are discussed. Time cycles and material balances are calculated.

9 1 1 2 3 4  
Kurath, D. E., 1985 (June), *Technology Study for the Pretreatment of Complexant Concentrate*, SD-WM-TA-010, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

Several alternatives for the treatment of transuranic (TRU) liquid wastes containing organic complexants were evaluated. TRU removal methods considered were TRUEX solvent extraction and co-precipitation by adding iron nitrate  $[\text{Fe}(\text{NO}_3)_3]$ . The TRUEX process uses an organic extractant to remove TRU and the  $\text{Fe}(\text{NO}_3)_3$  is thought to remove TRU by absorption. The TRU removal/organic destruction alternatives considered were ozonization, oxidation with hydrogen peroxide, high temperature/high pressure, and oxidation in supercritical water. These processes work by destroying the complexing ability of the organics, thereby allowing the TRU to precipitate. These alternatives were compared against the alternative of direct disposal of the waste in glass.

The TRUEX solvent extraction process was found to be the preferred alternative on the condition that the waste streams from this process are compatible with final disposal as grout and glass. The TRUEX process was found to minimize cost, maximize safety and utilize Hanford experience. The TRUEX process also has the flexibility to process other wastes such as existing Plutonium Finishing Plant waste and neutralized current acid waste. Extensive process development is required before this process can be implemented.

For organic destruction the most promising alternatives were found to be oxidation in supercritical water and oxidation with hydrogen peroxide. Extensive process development is required before these processes can be implemented."

9 1 1 2 3 4  
Kurath, D. E., 1986 (January), *Technology Program Plan for the Pretreatment of Complexant Concentrate*, SD-WM-TTP-018, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This technical plan describes the work effort to achieve the final disposal of complexant concentrate. Specifically this involves addressing technology for the following:

- Development of TRUEX solvent extraction for TRU removal from the complexant concentrate.
- Determine the need for organic destruction.
- Investigation of organic destruction methods as a contingency action.

Execution of this plan will be performed by Rockwell Hanford Operations, Oak Ridge National Laboratory, Argonne National Laboratory, and Pacific Northwest Laboratory.

Kurath, D. E., and C. J. Yeager, 1987 (May), *Integrated Technology Program Plan for the Treatment of NCRW*, SD-WM-TPP-036, Revision 1, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The NCRW sludge produced as of 1/1/87 has been shown to have a TRU radionuclide level that is too high for direct disposal of the sludge as a grout in near surface vaults. Direct vitrification is not desirable because of a large cost impact. Consequently, a pretreatment method or an alternative disposal method is required for the NCRW sludge. It is also desirable to find a process that can be implemented in PUREX that will produce a low-TRU NCRW sludge that is suitable for disposal in near surface vaults.

The proposed solutions for solving the problem in PUREX center around enhancing the solids removal capability. These methods are:

- addition of flocculating agents
- addition of alternative forms of rare earth
- inertial filtration
- pneumatic hydropulse filtration

The proposed solutions for dealing with the TRU NCRW sludge include:

- blending with other wastes for new surface grout disposal
- grout disposal at WIPP
- pretreatment to remove and concentrate the TRU fraction for vitrification and the low level fraction to near surface grout disposal

This technical program plan provides for the proper integration of chemical processing and waste management tasks to solve the TRU NCRW sludge problem. As part of this effort the following is included: task descriptions; status; cost estimates for unfunded tasks; organizations responsible for tasks; integrated schedule; and key technical decisions.

Lutton, T. W., W. W. Schulz, D. M. Strachan, and L. J. Bollyky, 1980 (March), *Ozonation of Hanford Nuclear Defense Waste*, RHO-SA-98, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

High (e.g., 0.1 to 0.5M) concentrations of ethylenediaminetetraacetate (EDTA), 2-hydroxyethylethylenediaminetriacetate (HEDTA), and other organic complexing agents are present in some of the nuclear waste

solutions currently stored at the U. S. Department of Energy Hanford Site in southeastern Washington State. Ozonolysis of these alkaline solutions smoothly and efficiently destroys the organic material thereby facilitating ion exchange removal of <sup>99</sup>Sr and other long-lived cationic radionuclides. Successful bench-scale ozonation tests have been performed with both synthetic and actual waste liquids.

Nankani, F. D., 1984 (October), *Hanford Waste Pretreatment Processes*, SD-RE-TI-134, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

The purpose of this document is to define the processes required to deliver an acceptable feed to the Hanford Waste Vittrification Plant (HWVP) for vittrification. This document includes the tanks available for storage, the use of existing equipment, and the necessary upgrades to existing equipment in B Plant. B Plant is the designated facility for all pretreatment processes.

These upgrades are necessary if B Plant is to pretreat the various Hanford Site wastes to make them suitable for immobilization to glass. The feed will be prepared from existing and future waste including neutralized current acid waste (NCAW), complexant concentrate (CC), and existing double-shell tank waste that requires vittrification.

The stream compositions and flow diagrams were developed to provide information to feed pretreatment, melter, and grout process design personnel.

Peterson, M. E., R. D. Scheele, and J. M. Tingey, 1989 (September), *Characterization of the First Core Sample of Neutralized Current Acid Waste from Double-Shell Tank 101-AZ*, PNL-xxxx, Pacific Northwest Laboratory, Richland, Washington.

Status or Availability (Checked): Public availability of document is being verified.

Richmond, W. G., 1988 (November 14), *Methods and Data for Use in Determining Source Terms for the Grout Disposal Program*, SD-WM-TI-355, Revision 1, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

RHO, 1980 (October), *Technical Status Report on Environmental Aspects of Long-Term Management of High-Level Defense Waste at the Hanford Site*, RHO-LD-139, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This report provides information on the environmental aspects of four alternative methods for long-term management of high-level defense radioactive wastes (HLW) stored at the U.S. Department of Energy (DOE) Hanford Site near Richland, Washington. This information will be used in preparing an environmental impact statement (EIS) on the disposition of Hanford defense waste (HDW). The HDW-EIS is planned for issuance in draft form for public comment and will also address the large amounts of transuranic (TRU) wastes at Hanford. However, this report addresses only the defense HLW.

Since 1944, radioactive wastes have accumulated at DOE's 500-km<sup>2</sup> (570-mi<sup>2</sup>) Hanford Site in southeastern Washington, where nine nuclear reactors have produced nuclear materials for National defense. Today, only one production reactor is still operating, but a large inventory of radioactive high-level waste, the residue from processing the spent fuel to recover plutonium and uranium, remains stored in underground tanks and in metal capsules in water basins. So that this waste will pose no significant threat to the public health and safety, it must be isolated from the biosphere for thousands of years.

This document contains an evaluation of environmental impacts of four alternative methods for long-term management of these HLW. The alternatives range from continuing the present action of storing the waste near the surface of the ground to retrieving the waste and disposing of it deep under ground in a mined geologic repository. The alternatives are:

- A - Near-term geologic disposal of stored waste
- B - Deferred geologic disposal of in-tank waste
- C - In situ disposal of in-tank waste
- D - Continued present action for stored waste

The environmental impacts of the four alternatives are small relative to that radiation received from natural sources or the available natural resources in the earth.

RHO, 1980 (October), *Technical Aspects of Long-Term Management Alternatives for High-Level Defense Waste at the Hanford Site*, RHO-LD-141, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This report provides information on technical aspects of nine alternative methods for long-term management of Hanford Site High-Level Wastes (HLW) (six for in-tank waste and three for encapsulated wastes).

The following conclusions can be drawn for in-tank waste:

- Continued storage in the tanks for an indefinite period of time, either with engineered barriers (in situ disposal, Specific Alternative 7) or without engineered barriers (Specific Alternative 8) is by far less costly than any alternative that requires processing the waste and disposing of it in another location. The next least expensive alternatives are those incorporating radionuclide concentration which greatly reduces the volume of waste requiring high integrity containerization and storage.

The following conclusion can be drawn for encapsulated waste:

- Continued storage in an onsite near-surface facility (Specific Alternative 11) is more costly than either alternative that requires disposal in a geologic repository due to the cost of surveillance for 250 years.

Routine and accidental releases of radionuclides were calculated and were largest for those alternatives in which in-tank wastes are retrieved and processed for disposal in a geologic repository.

On basis of results and insights gained during evaluation of the alternatives described in this report, guidance can be provided to narrow the scope of the technology development program as follows:

- Development efforts for near-term disposal alternatives should focus on processes which reduce the volume of waste to be placed in a repository.
- Research and development efforts should be emphasized in areas relating to in situ (near-surface) disposal of in-tank waste, e.g., the potential for leaving as much waste as possible in tanks to reduce costs and potential radiologic risk while meeting criteria for safe storage and disposal of chemical and radioactive wastes.

RHO, 1983 (December), *Hanford Waste Management Technology Plan*, RHO-WM-PL-9, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

RHO, 1985 (December), *Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the Hanford Defense Waste - Environmental Impact Statement*, RHO-RE-ST-30 P, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

This document provides the engineering bases for the development of the Hanford Defense Waste-Environmental Impact Statement. In compliance with the National Environmental Policy Act process and, more



specifically, the detailed scope prepared for the Hanford Defense Waste-Environmental Impact Statement, four waste disposal alternatives are identified: geologic disposal; in-place stabilization and disposal; continued storage (no disposal action); and the reference alternative. For each disposal alternative, the following six waste type classifications are addressed: existing tank waste, transuranic-contaminated soil sites (cribs and reverse wells), pre-1970 transuranic buried solid waste sites, retrievably stored and newly generated solid transuranic waste, strontium and cesium capsules, and future tank waste. The disposal alternatives are presented as options for the disposal of each waste type. Data regarding structures, site locations, and inventories for each waste class are provided, and are followed by a description of various technologies applied for implementing the disposal alternatives. Data associated with the resulting impacts (resources consumed, manpower used, emissions, and costs) are tabulated according to the waste class/alternative matrix. This information was used during the preparation of the Hanford Defense Waste-Environmental Impact Statement to develop socioeconomic analyses, accident scenarios, dose estimates, and waste release or migration evaluations.

RHO, 1986 (November), *Integrated Grout Management Plan*, RHO-RE-PL-12P, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

RHO, 1987 (February), *Engineering Support Data Update for the Hanford Defense Waste - Environmental Impact Statement*, RHO-RE-ST-30 ADD P, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

This document provides updated engineering support data for development of an environmental impact statement for Hanford defense, high-level, transuranic, and tank wastes. This document should be used in conjunction with the original engineering support data entitled Hanford Defense Waste - Environmental Impact Statement. The update data are intended to reflect data and information gathered since 1983, and are current to January 1987. Updated data include inventories, site descriptions, engineering methodologies for retrieval of single-shell tank waste, and facilities descriptions and costs. Errata for the original engineering data is also included as an appendix.

Richardson, G. L., 1980 (November), *Deferred Processing of Hanford High-Level Wastes*, HEDL-TME 80-48, Hanford Engineering Development Laboratory, Westinghouse Hanford Company, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

This document was prepared to provide detailed engineering and environmental information on possible alternatives that may be used

for deferred retrieval and disposal of high-level in-tank wastes derived from the nuclear defense program at Hanford.

An evaluation of the incentives for deferred processing of high-level waste (HLW) shows that the implementation costs (using trust fund annuity financing) and potential radiological health effects to the public decrease with time for the first 200 years of deferral but tend to level off after about 200 to 300 years. Thus, a deferral period of 250 years was selected for detailed evaluation of the deferred retrieval alternatives. At this time, the gamma activity will have decreased to the point that the waste can be handled and disposed of as a low-gamma-level transuranic (TRU) waste.

To parallel Rockwell's near-term retrieval and processing scenarios, other processing concepts involving both radionuclide concentration and bulk disposal were evaluated. A bulk fused salt process was selected as the reference process for Specific Alternative 5 for onsite disposal in a basalt repository, and a modified radionuclide concentration/vitrification (RC/V) process was selected for Specific Alternative 6 for offsite disposal in a bedded salt repository. These processing scenarios are considered to adequately bracket the range of impacts that may be incurred for deferred processing.

Rizzo, A. J., 1989 (External Letter to R. M. Bernero, U.S. Nuclear Regulatory Commission, Washington, D.C., March 6, 1989), U.S. Department of Energy, Richland Operations Office, Richland, Washington.

Status or Availability: Publicly available document.

The classification of the fraction of double-shell tank waste that will be disposed of near-surface as grout is addressed. An overall radionuclide material balance for all Hanford wastes is presented. Based on consideration of alternative separation processes, and to meet the suggested criteria of segregating the largest practical amount of activity, removal of the least 95% of the Cs-137 from complexant concentrate waste is proposed.

Scheele, R. D., and D. McCarthy, 1986 (May), *Characterization of Actual Zirflex Decladding Sludge*, PNL-xxxx, Prepared by Pacific Northwest Laboratory for Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Public availability of document is being verified.

Scheele, R. D., and M. E. Peterson, 1990 (January), *Results of the Characterization of Samples of Waste from Double-Shell Tank 102-SY*, PNL-xxxx, Pacific Northwest Laboratory, Richland, Washington.

Status or Availability: Public availability of document is being verified.

Schulz, W. W., 1980 (January), *Cyclohexanone Solvent Extraction of  $^{99}\text{TcO}_4$  From Alkaline Nuclear Waste Solutions*, RHO-SA-123, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Laboratory scale tests were performed to evaluate a solvent extraction process for removing  $\text{TcO}_4^-$  (pertechnetate anion) from Hanford alkaline waste solutions using cyclohexanone as the extractant. Distribution coefficients of  $\text{TcO}_4^-$  between aqueous alkaline nitrate feed and cyclohexanone are high enough to permit satisfactory countercurrent engineering-scale extraction of  $^{99}\text{Tc}$ . Technetium can be removed from cyclohexanone extracts by simply stripping with water, although phase disengaging problems were encountered during water stripping operations on a laboratory scale. Stripping tests in pulse columns and/or centrifugal contactors are needed to determine the magnitude of the phase disengaging problem and to find suitable remedies.

Schulz, W. W., 1980 (January), *Removal of Radionuclides from Hanford Defense Waste Solutions*, RHO-SA-51, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

The Hanford high-level defense wastes are characterized by their large volume ( $\sim 190\,000\text{ m}^3$ ) and varying content of inert and radioactive constituents. The water-soluble portion ( $\sim 140\,000\text{ m}^3$ ) of these wastes, which consists mainly of  $\text{NaNO}_3$ ,  $\text{NaAl}(\text{OH})_4$ ,  $\text{Na}_2\text{CO}_3$  and other sodium salts, contains a few milligrams of long-lived ( $t_{1/2} \geq 10$  years) radionuclides per 1000 kilograms. There is probable economic incentive for long-term management of Hanford defense wastes to partition them into a small volume of highly radioactive material requiring high integrity immobilization and storage and a much larger fraction of low-level (e.g.,  $< 10\text{ nCi/g}$ ) waste which can be economically and safely stored in bulk form. To aid in achieving this latter objective, an integrated series of aqueous separations processes (precipitation, ion exchange, and solvent extraction methods) was designed to remove  $^{137}\text{Cs}$ ,  $^{90}\text{Sr}$ , actinides, other multivalent cationic fission and activation products, and  $^{99}\text{Tc}$  from the water-soluble wastes. Results of generally satisfactory laboratory-scale tests of radionuclide removal technology with actual Hanford wastes are described.

Schulz, W. W., and L. D. McIsaac, 1975 (August), *Removal of Actinides from Nuclear Fuel Reprocessing Waste Solutions with Bidentate Organophosphorus Extractants*, ARH-SA-217, Atlantic Richfield Hanford Company, Richland, Washington.

Status or Availability (Checked): Publicly available document.

The neutral bidentate organophosphorus reagents DBDECMP (dibutyl-N,N-diethylcarbamylmethylenephosphonate) and its dihexyl analogue DHDECMP are candidate extractant for removal of actinides from certain acidic

waste streams produced at the U. S. Energy Research and Development Administration's Hanford and Idaho Falls sites. Various chemical and physical properties including availability, cost, purification, alpha radiolysis, and aqueous phase solubility of DBDECMP and DHDECMP are reviewed. A conceptual flowsheet employing a 15% DBDECMP (or DHDECMP)-CCl<sub>4</sub> extractant for removal (and recovery) of Am and Pu from Hanford's Plutonium Reclamation Facility acid waste stream (CAW solution) was successfully demonstrated in laboratory-scale mixer-settler tests; this extraction scheme can be used to produce an actinide-free waste. A 30% DBDECMP-xylene flowsheet is being tested at the Idaho Falls site for removal of U, Np, Pu, and Am from Idaho Chemical Processing Plant first-cycle high-level raffinate to produce an actinide-free (<10 nCi alpha activity/gram) waste.

Schulz, W. W., M. J. Kupfer, and J. P. Sloughter, 1983 (December), *Evaluation of Process and Facility Options for Treatment of Double-Shell Tank Wastes*, SD-WM-ES-023, Revision 0, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

An engineering study was performed to define and evaluate options for preparing existing and future double-shell tank wastes for immobilization (glass or grout); preferred feed preparation processes, facilities, and schedules were determined. Three preferred flowsheets for preparing immobilization facility feeds from six candidate wastes [Current Acid Wastes (CAW), Neutralized Current Acid Waste (NCAW), Double-Shell Slurry (DSS), Complexant Concentrate (CC), Cladding Removal Waste (CRW), and Plutonium Finishing Plant (PFP) Wastes] were derived by applying screening criteria to an initial 80 process options.

Three different facilities [B Plant, Expanded B Plant Immobilization Pilot Plant (BIPP) and New Stand Alone Facility] were evaluated for performing the waste preparation steps. Costs of conducting the preferred sequence of feed preparation operations in each of the three facilities were estimated for facility startup dates in the period 1986 to 2000.

Based upon analysis and evaluation of the significant findings of this study, the following facility selection and deployment schedule for feed preparation and immobilization facilities are recommended:

- Upgrade the existing B Plant for FY 1986 start of feed preparation operations for CRW, CC, and if necessary DSS wastes.
- Complete design and construction of a transportable grout facility to start immobilization and near-surface disposal of candidate wastes (DSS, Customer Wastes) in FY 1986.

- Complete design and construction of the BIPP facility to bring it on line in FY 1991.

Schulz, W. W., M. M. Beary, S. A. Gallagher, B. A. Higley, R. G. Johnston, F. M. Jungfleisch, M. J. Kupfer, R. A. Palmer, R. A. Watrous, and G. A. Wolf, 1980 (September), *Preliminary Evaluation of Alternative Forms for Immobilization of Hanford High-Level Defense Wastes*, RHO-ST-32, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

The document presents a preliminary evaluation of solid waste forms for immobilization of Hanford high-level radioactive defense wastes. Nineteen waste forms were evaluated and compared to determine their applicability and suitability for immobilization of Hanford salt cake, sludge, and residual liquid. Waste forms were evaluated and ranked on the basis of weighted ratings of seven waste form and seven process characteristics. Borosilicate glass was ranked among the first three choices for fixation of all Hanford HLW.

Schulz, W. W., M. M. Beary, R. A. Watrous, R. G. Johnston, and J. V. Panesko, 1982 (June), *Inventories and Technology for Recovery of Americium, Promethium, Rhodium, and Palladium Values at Hanford: A Preliminary Assessment*, RHO-LD-170, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Document is being cleared concurrently with WHC-EP-0365.

Inventories and current economic worths of  $^{241}\text{Am}$ ,  $^{147}\text{Pm}$ , and stable rhodium and palladium in existing Hanford wastes and in future Hanford PUREX Plant high-level waste were calculated and are presented. "Conceptual process flowsheets for recovery of a crude americium and promethium fraction either in the PUREX Plant (via DHDECMP solvent extraction) or in B Plant (via currently used HDEHP solvent extraction) are presented. A pyrometallurgical process under development at Pacific Northwest Laboratory appears technically suitable for recovery of a crude rhodium and palladium fraction from vitrified acidic (and possibly alkaline) future PUREX high-level waste. A conceptual amine solvent extraction process that involves head-end removal of aluminum and  $^{99}\text{Tc}$  is discussed for recovery of rhodium and palladium values from existing highly alkaline waste. Major research and development tasks needed to implement and/or support recovery of  $^{241}\text{Am}$ ,  $^{147}\text{Pm}$ , rhodium, and palladium from future PUREX high-level waste and/or existing wastes are listed.

Schwoebel, R. L., and C. J. Northrup, 1978 (November), *Proceedings of the Sandia Laboratories Workshop on the Use of Titanate Ion Exchangers for Defense Waste Management*, SAND78-2019, Sandia National Laboratory, Albuquerque, New Mexico.

Status or Availability (Checked): Publicly available document.

This workshop convened some of the principal technical participants involved in programs for the stabilization of tank-stored defense wastes at Savannah River Laboratories, the Hanford Reservation, and Nuclear Fuel Services at West Valley, New York. The purpose of the workshop was to discuss baseline objectives and decontamination processes currently planned and/or being investigated by each facility, review studies at Rockwell and Sandia Laboratories of the scientific and engineering applicability of a generic family of inorganic ion exchangers to waste decontamination, and identify future research and development activities required to implement use of these ion exchangers in full-scale decontamination.

The applicability of inorganic titanate ion exchangers to a wide variety of waste management applications, was reviewed. Since 1975, the research on these materials has been directed toward solving the problem of the defense waste decontamination. In a joint program with the Atlantic Richfield Hanford Company, Sandia Laboratories investigated the possibility of efficiently extracting the multivalent ions (principally  $^{90}\text{Sr}$ ) from the basic, high-salt content defense wastes. These studies indicate that Sr and actinides can be removed from salt cake to such an extent that the total residual activity is  $\sim 10 \text{ n Ci/gm}$ .

A feature of these exchangers that stimulated some interest were experiments indicating that the material could be efficiently eluted. Effective elution could significantly impact the flow sheets by decreasing costs, down time, and operational complexity. It was recommended that additional experiments be performed to detail the elution properties. It was also recommended that the ion exchange properties be investigated to determine the temperature interval over which this material can be processed and still retain its high affinity.

WHC, 1990 (January), *Assessment of Double-Shell Tank Waste Pretreatment Options*, WHC-SP-0464, Revision 1, Westinghouse Hanford Company, Richland, Washington.

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Some Hanford Site liquid and solid wastes stored in double-shell tanks (DST) must be pretreated before final disposal in cementitious grout or glass forms. The current baseline waste management plan calls for necessary pretreatment operations to be performed in the upgraded B Plant facility. In addition to the viability of B Plant for pretreatment of DST waste, a comprehensive consideration and examination of alternative facilities, including B Plant, for performing required pretreatment operations was made. A key step in evaluation of the options involved determination of the viability of the existing B Plant facility for the waste pretreatment mission, and the 244-AR Vault for waste lag storage and sludge washing operations.

No issues were found that would prevent B Plant or the 224-AR Vault from completing the pretreatment missions. The need for some additional facility upgrades was identified. With these upgrades the facilities can be brought to a condition that will comply with DOE design criteria, safety, and environmental orders.

Three alternative process and facility strategies were developed and compared. One option (Option B) with sludge washing in the 244-AR Vault and early TRUEX process operations in B Plant resulted in significant cost savings compared to the other options examined. Key studies were proposed to support final approval of the preferred strategy.

Winters, W. I., 1981 (June), *Effect of pH on the Destruction of Complexants with Ozone in Hanford Nuclear Waste*, RHO-SA-203, Rockwell Hanford Operations, Richland, Washington.

Status or Availability (Checked): Publicly available document.

Chemical processing of nuclear waste at Hanford has generated some waste solutions with high concentration (0.1 to 0.5M) of N-(hydroxyethyl)-ethylenediaminetriacetic acid (HEDTA), ethylenediaminetetraacetic acid (EDTA), and other organic complexing agents. These complexants must be destroyed because they affect radionuclide migration in soils, waste concentration, radionuclide removal, and other waste storage and processing considerations. Previous studies on actual waste solutions demonstrated that pre-ozonation of the alkaline waste significantly improved radionuclide removal. A series of bench-scale experiments using synthetic waste has been performed to determine the optimum pH for most efficient ozone destruction of EDTA. Ozonation of EDTA in synthetic waste was carried out over the pH range of 1 to 14. Potential catalytic materials were examined at different pH levels. The EDTA-ozone reaction rates and stoichiometric requirements were compared and evaluated for the varying conditions.

Wong, J. J., 1989 (October), *244-AR Conceptual Flowsheet for Processing of NCAW*, WHC-SE-WM-TI-396, Revision 0, Westinghouse Hanford Operations, Richland, Washington.

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